Pacific Gas and Electric Company®

James R. Becker
Vice President
Diablo Canyon Operations and
Station Director

Diablo Canyon Power Plant
P. O. Box 56
Avila Beach, CA 93424

805.545.3462
Fax: 805.545.4234

May 30, 2006

PG&E Letter DCL-06-061

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
License Amendment Request 06-04
Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity (TSTF-449)

Dear Commissioners and Staff:

In accordance with 10 CFR 50.90, enclosed is an application for amendment to Facility Operating License Nos. DPR-80 and DPR-82 for Units 1 and 2 of the Diablo Canyon Power Plant, respectively.

The license amendment request (LAR) proposes to revise Technical Specification (TS) requirements related to steam generator tube integrity. The change is consistent with NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity." The availability of this TS improvement was announced in the Federal Register on May 6, 2005 (70 FR 24126), as part of the consolidated line item improvement process.

Enclosure 1 contains a description of the proposed changes, the supporting technical analyses, and the no significant hazards consideration determination. Enclosures 2 and 3 contain marked-up and retyped (clean) TS pages, respectively. Enclosure 4 provides the marked-up TS Bases changes. TS Bases changes will be implemented pursuant to TS 5.5.14, "Technical Specifications Bases Control Program," at the time this amendment is implemented.

Pacific Gas and Electric Company (PG&E) has determined that this LAR does not involve a significant hazard consideration as determined per 10 CFR 50.92. Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of this amendment.

A001

The changes in this LAR are not required to address an immediate safety concern. PG&E requests the license amendment(s) be made effective upon NRC issuance, and to be implemented within 120 days of the date of issuance.

This communication contains no new commitments.

If you have any questions or require additional information, please contact Stan Ketelsen at 805-545-4720.

Sincerely,

James R. Becker
Vice President - Diablo Canyon Operations and Station Director

kjse/4328
Enclosures
cc: Edgar Bailey, DHS
Terry W. Jackson
Bruce S. Mallett
Diablo Distribution
cc/enc: Alan B. Wang

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

	)	Docket No. 50-275
In the Matter of	)	Facility Operating License
PACIFIC GAS AND ELECTRIC COMPANY	)	No. DPR-80
	)	
Diablo Canyon Power Plant	)	Docket No. 50-323
Units 1 and 2	)	Facility Operating License
	)	No. DPR-82

AFFIDAVIT

James R. Becker, of lawful age, first being duly sworn upon oath states that he is Vice President - Diablo Canyon Operations and Station Director of Pacific Gas and Electric Company; that he has executed License Amendment Request 06-04 on behalf of said company with full power and authority to do so; that he is familiar with the content thereof; and that the facts stated therein are true and correct to the best of his knowledge, information, and belief.

James R. Becker
Vice President - Diablo Canyon Operations and Station Director

Subscribed and sworn to before me this 30th day of May, 2006, by James R. Becker, personally known to me or proved to me on the basis of satisfactory evidence to be the person who appeared before me.

Karen T. Mason
Notary Public
County of San Luis Obispo
State of California

KAREN T. MASON
Commission # 1428454
Notary Public - California
San Luis Obispo County
My Comm. Expires Jul 31, 2007

EVALUATION

1.0 DESCRIPTION

This letter is a request to amend Operating Licenses DPR-80 and DPR-82 for Units 1 and 2 of the Diablo Canyon Power Plant (DCPP), respectively.

The proposed changes would revise the Operating Licenses to revise the requirements in Technical Specification (TS) related to steam generator tube integrity. The changes are consistent with NRC approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity," Revision 4. The availability of this technical specification improvement was announced in the Federal Register on May 6, 2005, as part of the consolidated line item improvement process (CLIIP).

2.0 PROPOSED CHANGES

Consistent with the NRC-approved Revision 4 of TSTF-449, the proposed TS changes include:

- Revised TS definition of LEAKAGE
- Revised TS 3.4.13, "RCS Operational Leakage"
- New TS 3.4.17, "Steam Generator (SG) Tube Integrity"
- Revised TS 5.5.9, "Steam Generator (SG) Tube Surveillance Program"
- Revised TS 5.6.10, "Steam Generator (SG) Tube Inspection Report"

Proposed revisions to the TS Bases are also included in this application. As discussed in the NRC's model safety evaluation, adoption of the revised TS Bases associated with TSTF-449, Revision 4, is an integral part of implementing this TS improvement. The changes to the affected TS Bases pages will be incorporated in accordance with the TS Bases Control Program.

The proposed TS changes are noted on the marked-up TS page provided in Enclosure 2. The proposed retyped TS is provided in Enclosure 3. The revised TS Bases is contained for information only in Enclosure 4.

3.0 BACKGROUND

The background for this application is adequately addressed by the NRC Notice of Availability published on May 6, 2005 (70 FR 24126), the NRC Notice for Comment published on March 2, 2005 (70 FR 10298), and TSTF-449, Revision 4.

4.0 REGULATORY REQUIREMENTS AND GUIDANCE

The applicable regulatory requirements and guidance associated with this application are adequately addressed by the NRC Notice of Availability published on May 6, 2005 (70 FR 24126), the NRC Notice for Comment published on March 2, 2005 (70 FR 10298), and TSTF-449, Revision 4.

5.0 TECHNICAL ANALYSIS

Pacific Gas and Electric Company (PG&E) has reviewed the safety evaluation (SE) published on March 2, 2005 (70 FR 10298), as part of the CLIIP Notice for Comment. This included the NRC staff's SE, the supporting information provided to support TSTF-449, and the changes associated with Revision 4 to TSTF-449. PG&E has concluded that the justifications presented in the TSTF proposal and the SE prepared by the NRC staff are applicable to DCPP, and justify this amendment for the incorporation of the changes to the DCPP TS.

In order to provide consistency with the new TSTF-449, TS 5.5.9.d, provisions for SG tube inspections which include the entire length of the SG tube, the current TS 5.5.9.d.1.k (ii) definition for the W-Star (W*) length is revised to delete the cold leg W* length, thereby excluding application of W* repair criteria to cold leg degradation. The current TS 5.5.9 W* repair criteria for DCPP do not include an NRC approved leak methodology for cold leg degradation. Therefore, the TS revision to delete the cold leg W* Length will ensure that all degradation left in service under the W* repair criteria will be supported by an NRC approved leak methodology.

6.0 REGULATORY ANALYSIS

A description of this proposed change and its relationship to applicable regulatory requirements and guidance was provided in the NRC Notice of Availability published on May 6, 2005 (70 FR 24126), the NRC Notice for Comment published on March 2, 2005 (70 FR 10298), and TSTF-449, Revision 4.

6.1 VERIFICATION AND COMMITMENTS

The following information is provided to support the NRC staff's review of this amendment application:

Diablo Canyon Power Plant, Units 1 and 2	
Steam Generator Model(s):	Westinghouse Model 51
Effective Full Power Years (EFPY) of service for currently installed SGs	Unit 1: 17.2 EFPY at end of cycle 13 (October 2005) Unit 2: Projected 17.4 EFPY at end of Cycle 13 (April 2006)
Tubing Material	Inconel 600 Mill annealed
Number of tubes per SG	3388
Number and percentage of tubes plugged in each SG	Unit 1 SG 1 - 232 tubes plugged (6.85%) Unit 1 SG 2 - 315 tubes plugged (9.30%) Unit 1 SG 3 - 92 tubes plugged (2.72%) Unit 1 SG 4 - 186 tubes plugged (5.49%) Unit 2 SG 1 - 125 tubes plugged (3.69%) Unit 2 SG 2 - 268 tubes plugged (7.91%) Unit 2 SG 3 - 130 tubes plugged (3.84%) Unit 2 SG 4 - 367 tubes plugged (10.8%)
Number of tubes repaired in each SG	None
Degradation mechanism(s) identified	• Axial outside diameter stress corrosion cracking (ODSCC) at tube support plate (TSP) intersections • Volumetric indications at TSP intersections • Several types of stress corrosion cracking at dented TSP intersections: Axial primary water stress corrosion cracking (PWSCC), axial ODSCC, circumferential PWSCC, circumferential ODSCC, combination axial PWSCC and axial ODSCC, combination axial and circumferential indications • Axial PWSCC in Row 1 and Row 2 U-bends • Circumferential PWSCC in Row 1 U-bends • Circumferential PWSCC in greater than Row 2 U-bends • Several types of Hot Leg Tubesheet Region Degradation: Axial ODSCC, axial PWSCC, circumferential PWSCC, circumferential ODSCC, volumetric indications

Diablo Canyon Power Plant, Units 1 and 2	
	• Axial PWSCC and circumferential PWSCC in Hot Leg Deplugged Tube Plug Expansion Zone • Axial ODSCC in free span dings • Cold Leg Thinning at cold leg TSPs • Anti-Vibration bar wear • Ligament cracking in TSPs
Current primary-to-secondary leakage limits:	Per SG: 150 gallons per day (gpd) through any one SG Leakage is evaluated at: Room temperature
Approved Alternate Tube Repair Criteria (ARC): 1. Voltage-Based Repair Criteria	Approved by: Amendment Nos. 124 (Unit 1) and 122 (Unit 2) dated 03/12/98. Applicability (e.g., degradation mechanism, location): Axial ODSCC within TSPs. Special limits on allowable accident leakage: None. Exceptions or clarifications to the structural performance criteria that apply to the ARC: None.
2. W* Criteria	Approved by: Amendment Nos. 182 (Unit 1) and 184 (Unit 2) dated 10/28/05. Applicability (e.g., degradation mechanism, location): Predominately axially oriented inside diameter stress corrosion cracking confined within the tubesheet, below the bottom of the WEXTEX transition; any combination of tube degradation below the flexible W* length is acceptable. Special limits on allowable accident leakage: None.

Diablo Canyon Power Plant, Units 1 and 2	
3. Axial PWSCC Depth-Based Repair Criteria	Exceptions or clarifications to the structural performance criteria that apply to the ARC: None. Approved by: Amendment Nos. 152 (Unit 1) and 152 (Unit 2) dated 05/01/02. Applicability (e.g., degradation mechanism, location): Axial PWSCC within the thickness of dented TSPs. Special limits on allowable accident leakage: None. Exceptions or clarifications to the structural performance criteria that apply to the ARC: None.
Approved SG Tube Repair Methods	None.
Performance criteria for accident leakage	Primary-to-secondary leak rate values assumed in licensing basis accident analysis, including assumed temperature conditions: The primary-to-secondary leak rate values assumed in the licensing basis main steam line break accident radiological consequences analysis is 10.5 gallons per minute (gpm) in the faulted SG and 150 gpd in each of the three intact SGs. These leak rates are at room temperature conditions. Performance criteria: For combined alternate repair criteria and non-alternate repair criteria degradation, accident induced leakage is not to exceed 10.5 gpm (at room temperature) in the faulted SG. For only non-alternate repair criteria degradation, accident induced leakage in the faulted SG is not to exceed 1 gpm (at hot conditions), or 0.72 gpm at room temperature conditions, because the accident analyses assume a pre-existing 1 gpm total leakage to all SGs prior to the accident.

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

PG&E has reviewed the proposed no significant hazards consideration determination published on March 2, 2005 (70 FR 10298), as part of the CLIIP. PG&E has concluded that the proposed determination presented in the notice is applicable to DCPP, and the determination is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91(a).

8.0 ENVIRONMENTAL EVALUATION

PG&E has reviewed the environmental evaluation included in the model SE published on March 2, 2005 (70 FR 10298), as part of the CLIIP. PG&E has concluded that the staff's findings presented in that evaluation are applicable to DCPP and the evaluation is hereby incorporated by reference for this application.

9.0 PRECEDENT

This application is being made in accordance with the CLIIP. PG&E is not proposing variations or deviations from the TS changes described in TSTF-449, Revision 4, or the NRC staff's model SE published on March 2, 2005 (70 FR 10298).

10.0 REFERENCES

Federal Register Notices:

Notice for Comment published on March 2, 2005 (70 FR 10298)
Notice of Availability published on May 6, 2005 (70 FR 24126)

Proposed Technical Specification Changes (marked-up)

primary to secondary

(primary to secondary LEAKAGE)

1.1 Definitions

LEAKAGE (continued)

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

b. <u>Unidentified LEAKAGE</u>

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

c. <u>Pressure Boundary LEAKAGE</u>

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.

MODE

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

OPERABLE—OPERABILITY

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

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:

a. Described in Chapter 14 of the FSAR;

b. Authorized under the provisions of 10 CFR 50.59; or

c. Otherwise approved by the Nuclear Regulatory Commission.

(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 SG 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 limit.	B.1 Be in MODE 3. AND	6 hours
	B.2 Be in MODE 5.	36 hours

* For MODES 3 and 4, if steam generator water samples indicate less than the minimum detectable activity of 5.0 E-7 microcuries/ml for principal gamma emitters, the leakage requirement of specification 3.4.13.d. may be considered met.

tR

2. Not applicable to primary to secondary LEAKAGE.

1.

S

SURVEILLANCE REQUIREMENTS

	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.
SR 3.4.13.3	Verify steam generator primary-to-secondary leakage equal to or less than allowable.	72 hours

Insert 1

2

Insert 2

Insert 3

There are no changes to the page.
Page included for information only.

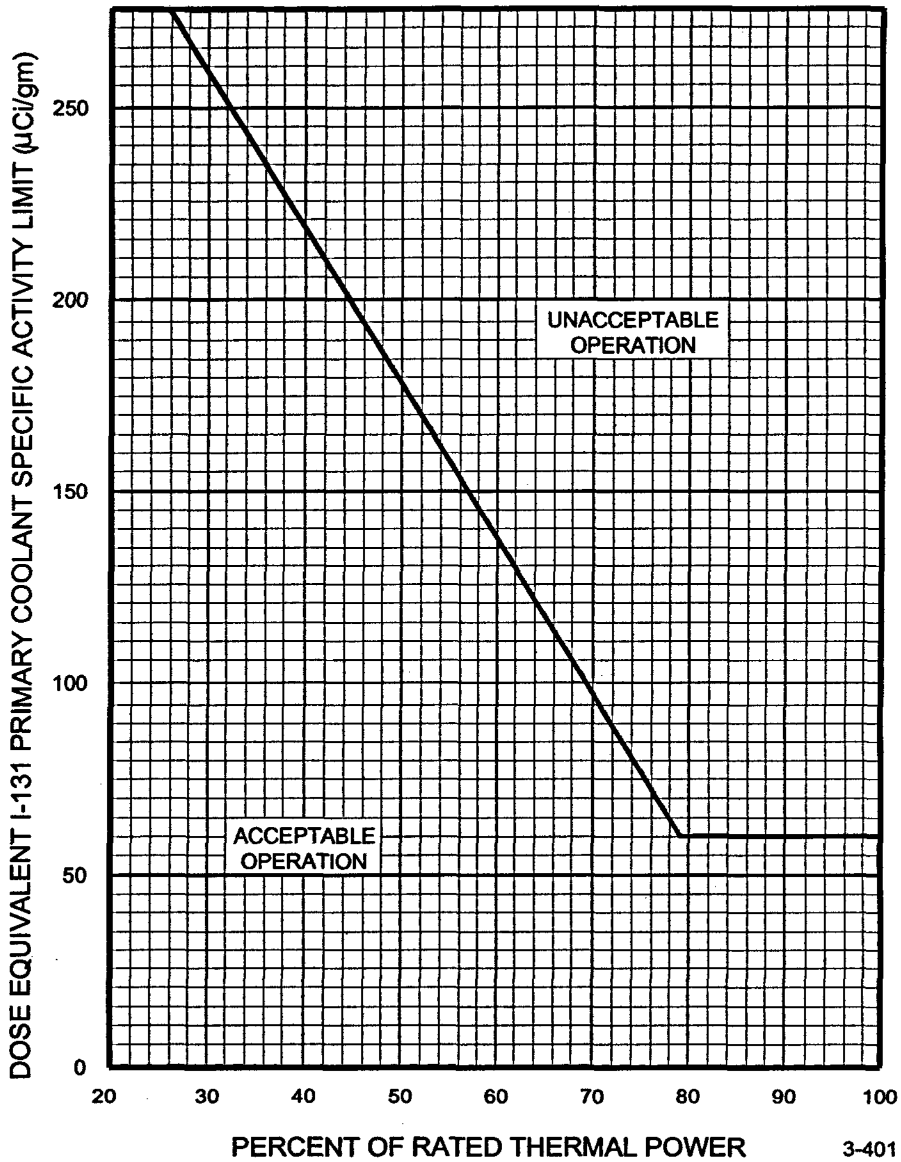

Figure 3.4.16-1

DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY > 1 µCI/GRAM DOSE EQUIVALENT I-131.

Insert Insert 4 for new TS 3.4.17, SG Tube Integrity, on next page

5.5 Programs and Manuals (continued) 9

5.5.9 Steam Generator (SG) Tube Surveillance Program

~~SG tube integrity shall be demonstrated by performance of the following augmented inservice inspection program.~~

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

~~a. SG Sample Selection and Inspection - SG tube integrity shall be determined during shutdown by selecting and inspecting at least the minimum number of SGs specified in Table 5.5.9-1.~~

~~b. SG Tube Sample Selection and Inspection - The SG tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.9-2. The inservice inspection of SG tubes shall be performed at the frequencies specified in Specification 5.5.9.c and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 5.5.9.d. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all SGs; the tubes selected for these inspections shall be selected on a random basis except:~~

Insert 5

Note, Insert 5 changes TS Section 5.5.9 in its entirety. The changes below identify current Section 5.5.9 text which is deleted or modified.

~~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 SG 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,~~

~~c) A tube inspection (pursuant to Specification 5.5.9.d.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,~~

9

Note, this is moved to 5.5.9.d.4

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

Note, this is moved to 5.5.9.d.5

e) Tubes identified as W* tubes having a previously identified indication within the flexible W* length shall be inspected using ~~a~~ n ~~rotating pancake coil (RPC)~~ probe or equivalent for the full length of the W* region during all future refueling outages.

(continued)

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

~~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 support plate intersections down to the lowest cold-leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold-leg tube support plate intersection having ODSCC indications shall be based on the performance of at least a 20% random sampling of tubes inspected over their full length.

Note, this is moved to 5.5.9.d.4

5. 6. Inspection of dented tube support plate intersections will be performed in accordance with WCAP-15573, Revision 1, to implement axial ~~primary water stress corrosion cracking (~~PWSCC~~)~~ depth-based repair criteria. The extent of required inspection is:

a) a. 100 percent bobbin coil inspection of all tube support plate (TSP) intersections.

b) b. Plus Point coil inspection of all bobbin coil indications at dented TSP intersections.

c) c. Plus Point coil inspection of all prior PWSCC indications left in service.

d) d. If bobbin coil is relied upon for detection of axial PWSCC in less than or equal to 2 volt dents, then on a SG basis perform Plus Point coil inspection of all TSP intersections having greater than 2 volt dents up to the highest TSP for which PWSCC has been detected in the prior two inspections or current inspection and 20% of greater than 2 volt dents at the next higher TSP. If a circumferential indication is detected in a dent of "x" volts in the prior two inspections or current inspection, Plus Point inspections will be conducted on 100% of dents greater than "x - 0.3" volts up to the affected TSP elevation in the affected SG, plus 20% of dents greater than "x – 0.3" volts at the next higher TSP. "x" is defined as the lowest dent voltage where a circumferential crack was detected.

Note, TS 5.5.9.b.5 is moved to 5.5.9.d.6

(continued)

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

~~e.~~ e) If bobbin coil is not relied upon for detection of axial PWSCC in less than or equal to 2 volt dents, then on a SG basis perform Plus Point coil inspection of all dented TSP intersections (no lower dent voltage threshold) up to the highest TSP for which PWSCC has been detected in the prior two inspections or current inspection and 20% of all dents at the next higher TSP.

~~f.~~ f) For any 20% dent sample, a minimum of 50 dents at the TSP elevation shall be inspected. If the population of dents is less than 50 at the TSP elevation, then 100% of the dents at the TSP elevation shall be inspected.

~~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.~~
~~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. Inspection Frequencies - The above required inservice inspections of SG tubes shall be performed at the following frequencies:~~

~~1. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections 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;~~

(continued)

Note, page number will change due to repagination of section 5.5.9 pages

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

~~2. If the results of the inservice inspection of a SG conducted in accordance with Table 5.5.9-2 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.c.1. The interval may then be extended to a maximum of once per 40 months; and~~

~~3. Additional, unscheduled inservice inspections shall be performed on each SG in accordance with the first sample inspection specified in Table 5.5.9-2 during the shutdown subsequent to any of the following conditions:~~

~~a) Reactor-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 Double Design 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.~~

~~d. 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. A tube containing a defect is defective;~~

~~f) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness.~~

~~1) This definition does not apply to tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to 5.5.9.d.1.j for the repair limit applicable to these intersections.~~

Note, page number will change due to repagination of section 5.5.9 pages

(continued)

The axial PWSCC depth-based repair criteria are used for disposition of

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

~~2) This definition does not apply to the portion of the tube within the tubesheet below the W* length. Acceptable tube wall degradation within the W* length shall be defined as in 5.5.9.d.1.k.~~

Note, this is moved to 5.5.9.c.3

3) ~~This definition does not apply to~~ axial PWSCC indications, or portions thereof, which are located within the thickness of dented tube support plates which exhibit a maximum depth greater than or equal to 40 percent of the initial tube wall thickness. WCAP-15573, Revision 1, provides repair limits applicable to these intersections.

Note, this is moved to 5.5.9.c.3

4) A tube which contains a tube support plate intersection with both an axial ODSCC indication and an axial PWSCC indication will be removed from service.

~~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 a Double Design Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.9.c.3, above;~~

Note, TS 5.5.9.d.1.h is moved to 5.5.9.d.5

h) ~~Tube Inspection means an~~ inspection of the ~~SG tube from the tube end (hot leg side) completely around the U-bend to the top support of the cold leg, excluding the portion of the tube within the~~ tubesheet ~~below the~~ Flexible W* Length or ~~below~~ 8 inches ~~from the hot leg top of tubesheet~~, ~~whichever is bounding;~~

,

.

below

~~i) Preservice Inspection means an inspection of the full length of each tube in each SG performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the field hydrostatic test and prior to initial Power Operation using the equipment and techniques expected to be used during subsequent inservice inspections;~~

The tube support plate voltage-based repair criteria are

j) ~~Tube Support Plate Plugging Limit~~ is used for the disposition of an alloy 600 steam generator tube for continued service that is experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the plugging limit is based on maintaining steam generator tube serviceability as described below:

Note, TS 5.5.9.d.1.j is moved to 5.5.9.c.1

a.

(i) 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 (NOTE 1), will be allowed to remain in service.

(continued)

Implementation of the W* repair criteria requires a 100 percent RPC probe or equivalent

Note, page number will change due to repagination of section 5.5.9 pages

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

5.5.9.c.1.c

b. (ii) Steam generator tubes, whose degradation is 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 (NOTE 1), will be repaired or plugged, except as noted in ~~5.5.9.d.1.j (iii)~~ below.

c. (iii) Steam generator tubes, with indication 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 (NOTE 1) but less than or equal to the upper voltage repair limit (NOTE 2), may remain in service 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 voltage repair limit (NOTE 2) will be plugged or repaired.

(continued)

Note, page number will change due to repagination of section 5.5.9 pages

5.5 Programs and Manuals

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

d.

(iv) Certain intersections as identified in PG&E Letter DCL-03-174, dated December 19, 2003, 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.

e.

(v) If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in 5.5.9.c.1.a, 5.5.9.c.1.b, and 5.5.9.c.1.c ~~5.5.9.d.1.j (i), 5.5.9.d.1.j (ii), and 5.5.9.d.1.j (iii)~~. The mid-cycle repair limits are determined from the following equations :

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

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

where :

V_{URL} = upper voltage repair limit

V_{LRL} = lower voltage repair limit

V_{MURL} = mid-cycle upper voltage repair limit based on time into cycle

V_{MLRL} = mid-cycle lower voltage repair limit based on V_{MURL} 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 rate per cycle length

NDE = 95% cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20% has been approved by the NRC)

Implementation of these mid-cycle repair limits should follow the same approach as in TS 5.5.9.c.1.a, 5.5.9.c.1.b, and 5.5.9.c.1.c ~~5.5.9.d.1.j (i), 5.5.9.d.1.j (ii), and 5.5.9.d.1.j (iii)~~.

(continued)

Note, page number will change due to repagination of section 5.5.9 pages

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

The W* repair criteria are

NOTE 1: The lower voltage repair limit is 2.0 volts for 7/8 inch diameter tubing at DCPP Units 1 and 2.

NOTE 2: The upper voltage repair limit is calculated according to the methodology in Generic Letter 95-05 as supplemented.

k) ~~W* Plugging Limit is~~ used for disposition of an alloy 600 steam generator tube for continued service that is experiencing predominately axially oriented inside diameter stress corrosion cracking confined within the tubesheet, below the bottom of the WEXTEX transition (BWT). As used in this specification:

Note, TS 5.5.9.d.1.k is moved to 5.5.9.c.2

a. (i) Bottom of WEXTEX Transition (BWT) is the highest point of contact between the tube and tubesheet at, or below the top-of-tubesheet as determined by eddy current testing.

b. (ii) W* Length is the distance in the tubesheet below the BWT that precludes tube pull out in the event of the complete circumferential separation of the tube below the W* length. The W* length is conservatively set at: 1) an undegraded hot leg tube length of 5.2 inches for Zone A tubes and 7.0 inches for Zone B tubes~~, and 2) an undegraded cold leg tube length of 5.5 inches for Zone A tubes and 7.5 inches for Zone B tubes~~. Information provided in WCAP-14797-P, Revision 2, defines the boundaries of Zone A and Zone B.

c. (iii) Flexible W* Length is the W* length adjusted for any cracks found within the W* region. The Flexible W* Length is the total ~~RPC~~ rotating pancake coil (RPC) inspected length as measured downward from the BWT, and includes NDE uncertainties and crack lengths within W* as adjusted for growth.

d. (iv) W* Tube is a tube with degradation within or below the W* length that is left in service, and degraded within the limits specified in Specification ~~5.5.9.d.1.k (v)~~ 5.5.9.c.2.e.

e. (v) Within the tubesheet, the plugging (repair) limit is based on maintaining steam generator serviceability as described below:

1. 1) For tubes to which the W* criteria are applied, the length of non-degraded tube below BWT shall be greater than or equal to the W* length plus NDE uncertainties and crack growth for the operating cycle.

(continued)

Note, page number will change due to repagination of section 5.5.9 pages

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

2. 2) Axial cracks in tubes returned to service using W* shall have the upper crack tip below the BWT by at least the NDE measurement uncertainty and crack growth allowance, such that at the end of the subsequent operating cycle the entire crack remains below the BWT.

3. 3) Resolvable, single axial indications (multiple indications must return to the null point between individual cracks) within the flexible W* length can be left in service. Alternate RPC coils or an ultrasonic test (UT) inspection can be used to demonstrate return to null point between multiple axial indications or the absence of circumferential involvement between axial indications.

4. 4) Tubes with inclined axial indications less than 2.0 inches long (including the crack growth allowance) having inclination angles relative to the tube axis of < 45 degrees minus the NDE uncertainty, ΔNDE_{CA}, on the measurement of the crack angle can be left in service. Tubes with two or more parallel (overlapping elevation), inclined axial cracks shall be plugged or repaired. For application of the 2.0 inch limit, an inclined indication is an axial crack that is visually inclined on the RCP C-scan, such that an angular measurement is required, and the measured angle exceeds the measurement uncertainty of ΔNDE_{CA}.

5. 5) Circumferential, volumetric, and axial indications with inclination angles greater than (45 degrees - ΔNDE_{CA}) within the flexible W* length shall be plugged or repaired.

6. 6) Any type or combination of tube degradation below the flexible W* length is acceptable.

~~2. The SG tube integrity shall be determined after completing the corresponding actions (plug all tubes exceeding the plugging limit) required by Table 5.5.9-2.~~

~~e. Reports~~

~~The contents and frequency of reports concerning the SG tube surveillance program shall be in accordance with Specification 5.6.10.~~

(continued)

Note, page number will change due to repagination of section 5.5.9 pages

THIS PAGE NOT USED

Note, page number will change due to repagination of section 5.5.9 pages

9

5.5 Programs and Manuals (continued)

~~TABLE 5.5.9-1~~
~~MINIMUM NUMBER OF STEAM GENERATORS (SGs) TO BE INSPECTED DURING INSERVICE INSPECTION~~

~~Preservice Inspection~~	~~No~~			~~Yes~~		
~~No. of Steam Generators per Unit~~	~~Two~~	~~Three~~	~~Four~~	~~Two~~	~~Three~~	~~Four~~
~~First Inservice Inspection~~	~~All~~			~~One~~	~~Two~~	~~Two~~
~~Second & Subsequent Inservice Inspections~~	~~One~~[1]			~~One~~[1]	~~One~~[2]	~~One~~[3]

~~TABLE NOTATIONS~~

~~1. The inservice inspection may be limited to one SG on a rotating schedule encompassing 3 N % of the tubes (where N is the number of SGs in the plant) if the results of the first or previous inspections indicate that all SGs are performing in a like manner. Note that under some circumstances, the operating conditions in one or more SGs may be found to be more severe than those in other SGs. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.~~

~~2. The other SG not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.~~

~~3. Each of the other two SGs not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.~~

(continued)

Note, page number will change due to repagination of section 5.5.9 pages

9

5.5 Programs and Manuals (continued)

~~TABLE 5.5.9-2~~
~~STEAM GENERATOR (SG) 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 defective tubes and inspect additional 2S tubes in this S.G.~~	~~C-1~~	~~None~~	~~N.A.~~	~~N.A.~~
			~~C-2~~	~~Plug defective tubes and inspect additional 4S tubes in this S.G.~~	~~C-1~~	~~None~~
					~~C-2~~	~~Plug defective tubes~~
					~~C-3~~	~~Perform action for C-3 result of first sample~~
	~~C-3~~	~~Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G.~~ ~~Notification to NRC pursuant to §50.72 (b) (2) of 10 CFR Part 50~~	~~C-3~~	~~Perform action for C-3 result of first sample~~	~~N.A.~~	~~N.A.~~
			~~All other S.G.s are C-1~~	~~None~~	~~N.A.~~	~~N.A.~~
			~~Some S.G.s C-2 but no additional S.G. are C-3~~	~~Perform action for C-2 result of second sample~~	~~N.A.~~	~~N.A.~~
			~~Additional S.G. is C-3~~	~~Inspect all tubes in each S.G. and plug defective tubes. Notification to NRC pursuant to §50.72 (b) (2) of 10 CFR Part 50~~	~~N.A.~~	~~N.A.~~

$\cancel{S = 3\frac{N}{n}\%}$ ~~Where N is the number of SGs in the unit, and n is the number of SGs inspected during an inspection~~

(continued)

Note, page number will change due to repagination of section 5.5.9 pages

5.6.10 Steam Generator (SG) Tube Inspection Report

Insert 6

~~a. Within 15 days following the completion of each inservice inspection of SG tubes, the number of tubes plugged in each SG shall be reported to the Commission.~~

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

~~1) Number and extent of tubes inspected,~~

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

~~3) Identification of tubes plugged.~~

~~c. Results of SG tube inspections, which fall into Category C-3, shall be reported in a Special Report to the Commission 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.~~

b. ~~d.~~ For implementation of the voltage-based repair criteria to tube support plate intersections, notify the NRC prior to returning the steam generators to service should any of the following 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, increased by estimated leakage by all other sources (alternate repair criteria and non-alternate repair criteria indications), exceeds the leak limit determined from the licensing basis dose calculation for the postulated main steamline break for the next operating cycle.

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

3. 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.

(continued)

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

~~e.~~ c. The results of the inspection of W* tubes shall be reported to the Commission pursuant to 10 CFR 50.4 within 90 days following return to service of the steam generators. This report shall include:

~~1)~~ 1. Identification of W* tube indications and indications that do not meet W* requirements and were plugged or repaired, including the following information: the number of indications, the locations of the indications (relative to the BWT and TTS), the orientation (axial, circumferential, volumetric, inclined), the radial position of the tube within the tubesheet, the W* Zone of the tube, the severity of each indication (estimated depth), the side of the tube in which the indication initiated (inside or outside diameter), the W* inspection distance measured with respect to the BWT or TTS (whichever is lower), the length of axial indications, the angle of inclination of clearly skewed axial cracks (if applicable), verification that the upper crack tip of W* indications returned to service in the prior cycle remain below the BWT by at least the 95% confidence NDE uncertainty on locating the crack tip relative to the BWT, updated 95% growth rate for use in operational assessment, the cumulative number of indications detected in the tubesheet region as a function of elevation within the tubesheet, and the condition monitoring and operational assessment main steamline break leak rate for each indication and each SG in accordance with the leak rate methodology described in PG&E Letter DCL-05-018, dated March 11, 2005, as supplemented by PG&E Letter DCL-05-090, dated August 25, 2005.

~~2)~~ 2. Assessment of whether the results were consistent with expectations and, if not consistent, a description of the proposed corrective action.

~~f.~~ d. The aggregate calculated steam line break leakage from application of all alternate repair criteria and non-alternate repair criteria shall be reported to the Commission pursuant to 10 CFR 50.4 within 90 days following return to service of the steam generators.

~~g.~~ e. For implementation of the repair criteria for axial PWSCC at dented TSPs, the NRC shall be notified prior to startup, pursuant to 10CFR50.72, of the following conditions that indicate a failure of performance criteria:

~~1)~~ 1. The calculated SG probability of burst for condition monitoring exceeds 1×10^{-2}.

~~2)~~ 2. The calculated SG leakage for condition monitoring from all sources (all alternate repair criteria and non-alternate repair criteria indications) exceeds the leakage limit determined from the licensing basis steam line break dose calculation.

(continued)

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

~~h.~~ f. For implementation of the repair criteria for axial PWSCC at dented TSPs, the results of the condition monitoring and operational assessments will be reported to the NRC within 120 days following completion of the inspection. The report will include:

~~1)~~ 1. Tabulations of indications found in the inspection, tubes repaired, and tubes left in service under the ARC.

~~2)~~ 2. Growth rate distributions for indications found in the inspection and growth rate distributions used to establish the tube repair limits.

~~3)~~ 3. Plus Point confirmation rates for bobbin detected indications when bobbin is relied upon for detection of axial PWSCC in less than or equal to 2 volt dents.

~~4)~~ 4. For condition monitoring, an evaluation of any indications that satisfy burst margin requirements based on the Westinghouse burst pressure model, but do not satisfy burst margin requirements based on the combined ANL ligament tearing and throughwall burst pressure model.

(continued)

5.6 Reporting Requirements

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

5. 5) Performance evaluation of the operational assessment methodology for predicting flaw distributions as a function of flaw size.

6. 6) Evaluation results of number and size of previously reported versus new PWSCC indications found in the inspection, and the potential need to account for new indications in the operational assessment burst evaluation.

7. 7) Identification of mixed mode (axial PWSCC and circumferential) indications found in the inspection and an evaluation of the mixed mode indications for potential impact on the axial indication burst pressures or leakage.

8. 8) Any corrective actions found necessary in the event that condition monitoring requirements are not met.

g. i. For implementation of the probability of prior cycle detection (POPCD) method, for the voltage-based repair criteria at tube support plate intersections, if the end-of-cycle conditional main steamline break burst probability, the projected main steamline break leak rate, or the number of indications are underpredicted by the previous cycle operational assessment, the following shall be reported to the Commission pursuant to 10 CFR 50.4 within 90 days following return to service of the steam generators:

1. 1) The assessment of the probable causes for the underpredications, proposed corrective actions, and any recommended changes to probability of detection or growth methodology indicated by potential methods assessments.

2. 2) An assessment of the potential need to revise the alternate repair criteria analysis methods if: the burst probability is underpredicted by more than 0.001 (i.e., 10% of the reporting threshold) or an order of magnitude; or the leak rate is underpredicted by more than 0.5 gpm or an order of magnitude.

3. 3) An assessment of the potential need to increase the number of predicted low voltage indications at the beginning of cycle if the total number of as-found indications in any SG are underestimated by greater than 15% or by greater than 150 indications.

Technical Specification Inserts

Insert 1

Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance.

Insert 2

Verify primary to secondary LEAKAGE is $\leq$ 150 gallons per day through any one SG.

Insert 3

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

Insert 4

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 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 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 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. AND B.2 Be in MODE 5.	6 hours 36 hours

Technical Specification Inserts (continued)

Insert 4 (continued)

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 in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

Insert 5

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

Insert 5 (continued)

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 for the faulted SG, except for specific types of degradation at specific locations as described in paragraph c of the Steam Generator Program in which leakage from application of all alternate tube repair criteria degradation and non-alternate tube repair criteria degradation is not to exceed 10.5 gpm.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria.

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

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

1. Tube Support Plate Voltage-Based Repair Criteria

The tube support plate voltage-based repair criteria are used for the disposition of an alloy 600 steam generator tube for continued service that is experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the plugging limit is based on maintaining steam generator tube serviceability as described below:

a. 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 (NOTE 1), will be allowed to remain in service.

b. Steam generator tubes, whose degradation is 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 (NOTE 1), will be repaired or plugged, except as noted in 5.5.9.c.1.c below.

Insert 5 (continued)

c. Steam generator tubes, with indication 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 (NOTE 1) but less than or equal to the upper voltage repair limit (NOTE 2), may remain in service 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 voltage repair limit (NOTE 2) will be plugged or repaired.

d. Certain intersections as identified in PG&E Letter DCL-03-174, dated December 19, 2003, 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.

e. If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in 5.5.9.c.1.a, 5.5.9.c.1.b, and 5.5.9.c.1.c. The mid-cycle repair limits are determined from the following equations :

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

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

where:

V_{URL} = upper voltage repair limit

V_{LRL} = lower voltage repair limit

V_{MURL} = mid-cycle upper voltage repair limit based on time into cycle

V_{MLRL} = mid-cycle lower voltage repair limit based on V_{MURL} 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 rate per cycle length

NDE = 95% cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20% has been approved by the NRC)

Insert 5 (continued)

Implementation of these mid-cycle repair limits should follow the same approach as in TS 5.5.9.c.1.a, 5.5.9.c.1.b, and 5.5.9.c.1.c.

NOTE 1: The lower voltage repair limit is 2.0 volts for 7/8 inch diameter tubing at DCPP Units 1 and 2.

NOTE 2: The upper voltage repair limit is calculated according to the methodology in Generic Letter 95-05 as supplemented.

2. W* Repair Criteria

The W* repair criteria are used for disposition of an alloy 600 steam generator tube for continued service that is experiencing predominately axially oriented inside diameter stress corrosion cracking confined within the tubesheet, below the bottom of the WEXTEX transition (BWT). As used in this specification:

a. Bottom of WEXTEX Transition (BWT) is the highest point of contact between the tube and tubesheet at, or below the top-of-tubesheet as determined by eddy current testing.

b. W* Length is the distance in the tubesheet below the BWT that precludes tube pull out in the event of the complete circumferential separation of the tube below the W* length. The W* length is conservatively set at an undegraded hot leg tube length of 5.2 inches for Zone A tubes and 7.0 inches for Zone B tubes. Information provided in WCAP-14797-P, Revision 2, defines the boundaries of Zone A and Zone B.

c. Flexible W* Length is the W* length adjusted for any cracks found within the W* region. The Flexible W* Length is the total rotating pancake coil (RPC) inspected length as measured downward from the BWT, and includes NDE uncertainties and crack lengths within W* as adjusted for growth.

d. W* Tube is a tube with degradation within or below the W* length that is left in service, and degraded within the limits specified in Specification 5.5.9.c.2.e.

e. Within the tubesheet, the plugging (repair) limit is based on maintaining steam generator serviceability as described below:

   1. For tubes to which the W* criteria are applied, the length of non-degraded tube below BWT shall be greater than or equal to the W* length plus NDE uncertainties and crack growth for the operating cycle.

   2. Axial cracks in tubes returned to service using W* shall have the upper crack tip below the BWT by at least the NDE measurement uncertainty and crack growth allowance, such that at the end of the subsequent operating cycle the entire crack remains below the BWT.

Insert 5 (continued)

3. Resolvable, single axial indications (multiple indications must return to the null point between individual cracks) within the flexible W* length can be left in service. Alternate RPC coils or an ultrasonic test (UT) inspection can be used to demonstrate return to null point between multiple axial indications or the absence of circumferential involvement between axial indications.

4. Tubes with inclined axial indications less than 2.0 inches long (including the crack growth allowance) having inclination angles relative to the tube axis of < 45 degrees minus the NDE uncertainty, ΔNDE_{CA}, on the measurement of the crack angle can be left in service. Tubes with two or more parallel (overlapping elevation), inclined axial cracks shall be plugged or repaired. For application of the 2.0 inch limit, an inclined indication is an axial crack that is visually inclined on the RCP C-scan, such that an angular measurement is required, and the measured angle exceeds the measurement uncertainty of ΔNDE_{CA}.

5. Circumferential, volumetric, and axial indications with inclination angles greater than (45 degrees - ΔNDE_{CA}) within the flexible W* length shall be plugged or repaired.

6. Any type or combination of tube degradation below the flexible W* length is acceptable.

3. Axial Primary Water Stress Corrosion Cracking (PWSCC) Depth-Based Repair Criteria

The axial PWSCC depth-based repair criteria are used for disposition of axial PWSCC indications, or portions thereof, which are located within the thickness of dented tube support plates which exhibit a maximum depth greater than or equal to 40 percent of the initial tube wall thickness. WCAP-15573, Revision 1, provides repair limits applicable to these intersections.

A tube which contains a tube support plate intersection with both an axial ODSCC indication and an axial PWSCC indication will be removed from service.

Technical Specification Inserts (continued)

Insert 5 (continued)

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, d.3, d.4, d.5, and d.6 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.

2. Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.

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

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

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

Technical Specification Inserts (continued)

Insert 5 (continued)

5. Tubes identified as W* tubes having a previously identified indication within the flexible W* length shall be inspected using an RPC probe or equivalent for the full length of the W* region during all future refueling outages.

   Implementation of the W* repair criteria requires a 100 percent RPC probe or equivalent inspection of the hot leg tubesheet Flexible W* Length, or 8 inches below the hot leg top of tubesheet, whichever is bounding.

6. Inspection of dented tube support plate intersections will be performed in accordance with WCAP-15573, Revision 1, to implement axial PWSCC depth-based repair criteria. The extent of required inspection is:

   a. 100 percent bobbin coil inspection of all tube support plate (TSP) intersections.

   b. Plus Point coil inspection of all bobbin coil indications at dented TSP intersections.

   c. Plus Point coil inspection of all prior PWSCC indications left in service.

   d. If bobbin coil is relied upon for detection of axial PWSCC in less than or equal to 2 volt dents, then on a SG basis perform Plus Point coil inspection of all TSP intersections having greater than 2 volt dents up to the highest TSP for which PWSCC has been detected in the prior two inspections or current inspection and 20% of greater than 2 volt dents at the next higher TSP. If a circumferential indication is detected in a dent of "x" volts in the prior two inspections or current inspection, Plus Point inspections will be conducted on 100% of dents greater than "x - 0.3" volts up to the affected TSP elevation in the affected SG, plus 20% of dents greater than "x – 0.3" volts at the next higher TSP. "x" is defined as the lowest dent voltage where a circumferential crack was detected.

   e. If bobbin coil is not relied upon for detection of axial PWSCC in less than or equal to 2 volt dents, then on a SG basis perform Plus Point coil inspection of all dented TSP intersections (no lower dent voltage threshold) up to the highest TSP for which PWSCC has been detected in the prior two inspections or current inspection and 20% of all dents at the next higher TSP.

   f. For any 20% dent sample, a minimum of 50 dents at the TSP elevation shall be inspected. If the population of dents is less than 50 at the TSP elevation, then 100% of the dents at the TSP elevation shall be inspected.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

Insert 6

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

1. The scope of inspections performed on each SG,
2. Active degradation mechanisms found,
3. Nondestructive examination techniques utilized for each degradation mechanism,
4. Location, orientation (if linear), and measured sizes (if available) of service induced indications,
5. Number of tubes plugged during the inspection outage for each active degradation mechanism,
6. Total number and percentage of tubes plugged to date, and
7. The results of condition monitoring, including the results of tube pulls and in-situ testing.

Proposed Technical Specification Changes (retyped)

Remove Page	Insert Page
1.1-4	1.1-4
3.4-27	3.4-27
3.4-28	3.4-28
	3.4-38
	3.4-39
5.0-10	5.0-10
5.0-11	5.0-11
5.0-11a	5.0-12
5.0-12	5.0-13
5.0-13	5.0-14
5.0-13a	5.0-15
5.0-14	5.0-16
5.0-15	5.0-17
5.0-16	5.0-18
5.0-17	5.0-19
5.0-18	
5.0-19	
5.0-29	5.0-29
5.0-30	5.0-30
5.0-30a	5.0-30a
5.0-30b	5.0-30b

1.1 Definitions

LEAKAGE (continued)	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 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.
MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE—OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14 of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.

(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 limit.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

* For MODES 3 and 4, if steam generator water samples indicate less than the minimum detectable activity of 5.0 E-7 microcuries/ml for principal gamma emitters, the leakage requirement of specification 3.4.13.d. may be considered met.

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.

AND

All SG tubes satisfying the tube repair criteria shall be plugged 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 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 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

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 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

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 for the faulted SG, except for specific types of degradation at specific locations as described in paragraph c of the Steam Generator Program in which leakage from application of all alternate tube repair criteria degradation and non-alternate tube repair criteria degradation is not to exceed 10.5 gpm.

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria.

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

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

1. Tube Support Plate Voltage-Based Repair Criteria

The tube support plate voltage-based repair criteria are used for the disposition of an alloy 600 steam generator tube for continued service that is experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the plugging limit is based on maintaining steam generator tube serviceability as described below:

a. 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 (NOTE 1), will be allowed to remain in service.

b. Steam generator tubes, whose degradation is 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 (NOTE 1), will be repaired or plugged, except as noted in 5.5.9.c.1.c below.

c. Steam generator tubes, with indication 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 (NOTE 1) but less than or equal to the upper voltage repair limit (NOTE 2), may remain in service 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 voltage repair limit (NOTE 2) will be plugged or repaired.

d. Certain intersections as identified in PG&E Letter DCL-03-174, dated December 19, 2003, 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.

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

e. If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in 5.5.9.c.1.a, 5.5.9.c.1.b, and 5.5.9.c.1.c. The mid-cycle repair limits are determined from the following equations:

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

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

where:

V_{URL} = upper voltage repair limit

V_{LRL} = lower voltage repair limit

V_{MURL} = mid-cycle upper voltage repair limit based on time into cycle

V_{MLRL} = mid-cycle lower voltage repair limit based on V_{MURL} 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 rate per cycle length

NDE = 95% cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20% has been approved by the NRC)

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

Implementation of these mid-cycle repair limits should follow the same approach as in TS 5.5.9.c.1.a, 5.5.9.c.1.b, and 5.5.9.c.1.c.

NOTE 1: The lower voltage repair limit is 2.0 volts for 7/8 inch diameter tubing at DCPP Units 1 and 2.

NOTE 2: The upper voltage repair limit is calculated according to the methodology in Generic Letter 95-05 as supplemented.

2. W* Repair Criteria

The W* repair criteria are used for disposition of an alloy 600 steam generator tube for continued service that is experiencing predominately axially oriented inside diameter stress corrosion cracking confined within the tubesheet, below the bottom of the WEXTEX transition (BWT). As used in this specification:

a. Bottom of WEXTEX Transition (BWT) is the highest point of contact between the tube and tubesheet at, or below the top-of-tubesheet as determined by eddy current testing.

b. W* Length is the distance in the tubesheet below the BWT that precludes tube pull out in the event of the complete circumferential separation of the tube below the W* length. The W* length is conservatively set at an undegraded hot leg tube length of 5.2 inches for Zone A tubes and 7.0 inches for Zone B tubes. Information provided in WCAP-14797-P, Revision 2, defines the boundaries of Zone A and Zone B.

c. Flexible W* Length is the W* length adjusted for any cracks found within the W* region. The Flexible W* Length is the total rotating pancake coil (RPC) inspected length as measured downward from the BWT, and includes NDE uncertainties and crack lengths within W* as adjusted for growth.

d. W* Tube is a tube with degradation within or below the W* length that is left in service, and degraded within the limits specified in Specification 5.5.9.c.2.e.

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

e. Within the tubesheet, the plugging (repair) limit is based on maintaining steam generator serviceability as described below:

1. For tubes to which the W* criteria are applied, the length of non-degraded tube below BWT shall be greater than or equal to the W* length plus NDE uncertainties and crack growth for the operating cycle.

2. Axial cracks in tubes returned to service using W* shall have the upper crack tip below the BWT by at least the NDE measurement uncertainty and crack growth allowance, such that at the end of the subsequent operating cycle the entire crack remains below the BWT.

3. Resolvable, single axial indications (multiple indications must return to the null point between individual cracks) within the flexible W* length can be left in service. Alternate RPC coils or an ultrasonic test (UT) inspection can be used to demonstrate return to null point between multiple axial indications or the absence of circumferential involvement between axial indications.

4. Tubes with inclined axial indications less than 2.0 inches long (including the crack growth allowance) having inclination angles relative to the tube axis of < 45 degrees minus the NDE uncertainty, ΔNDE_{CA}, on the measurement of the crack angle can be left in service. Tubes with two or more parallel (overlapping elevation), inclined axial cracks shall be plugged or repaired. For application of the 2.0 inch limit, an inclined indication is an axial crack that is visually inclined on the RCP C-scan, such that an angular measurement is required, and the measured angle exceeds the measurement uncertainty of ΔNDE_{CA}.

5. Circumferential, volumetric, and axial indications with inclination angles greater than (45 degrees - ΔNDE_{CA}) within the flexible W* length shall be plugged or repaired.

6. Any type or combination of tube degradation below the flexible W* length is acceptable.

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

3. Axial Primary Water Stress Corrosion Cracking (PWSCC) Depth-Based Repair Criteria

The axial PWSCC depth-based repair criteria are used for disposition of axial PWSCC indications, or portions thereof, which are located within the thickness of dented tube support plates which exhibit a maximum depth greater than or equal to 40 percent of the initial tube wall thickness. WCAP-15573, Revision 1, provides repair limits applicable to these intersections.

A tube which contains a tube support plate intersection with both an axial ODSCC indication and an axial PWSCC indication will be removed from service.

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, d.3, d.4, d.5, and d.6 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.

2. Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.

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

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

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

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

5. Tubes identified as W* tubes having a previously identified indication within the flexible W* length shall be inspected using an RPC probe or equivalent for the full length of the W* region during all future refueling outages.

   Implementation of the W* repair criteria requires a 100 percent RPC probe or equivalent inspection of the hot leg tubesheet Flexible W* Length, or 8 inches below the hot leg top of tubesheet, whichever is bounding.

6. Inspection of dented tube support plate intersections will be performed in accordance with WCAP-15573, Revision 1, to implement axial PWSCC depth-based repair criteria. The extent of required inspection is:

   a. 100 percent bobbin coil inspection of all tube support plate (TSP) intersections.

   b. Plus Point coil inspection of all bobbin coil indications at dented TSP intersections.

   c. Plus Point coil inspection of all prior PWSCC indications left in service.

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

d. If bobbin coil is relied upon for detection of axial PWSCC in less than or equal to 2 volt dents, then on a SG basis perform Plus Point coil inspection of all TSP intersections having greater than 2 volt dents up to the highest TSP for which PWSCC has been detected in the prior two inspections or current inspection and 20% of greater than 2 volt dents at the next higher TSP. If a circumferential indication is detected in a dent of "x" volts in the prior two inspections or current inspection, Plus Point inspections will be conducted on 100% of dents greater than "x - 0.3" volts up to the affected TSP elevation in the affected SG, plus 20% of dents greater than "x – 0.3" volts at the next higher TSP. "x" is defined as the lowest dent voltage where a circumferential crack was detected.

e. If bobbin coil is not relied upon for detection of axial PWSCC in less than or equal to 2 volt dents, then on a SG basis perform Plus Point coil inspection of all dented TSP intersections (no lower dent voltage threshold) up to the highest TSP for which PWSCC has been detected in the prior two inspections or current inspection and 20% of all dents at the next higher TSP.

f. For any 20% dent sample, a minimum of 50 dents at the TSP elevation shall be inspected. If the population of dents is less than 50 at the TSP elevation, then 100% of the dents at the TSP elevation shall be inspected.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

THIS PAGE NOT USED

THIS PAGE NOT USED

5.6 Reporting Requirements (continued)

5.6.10 Steam Generator (SG) Tube Inspection Report

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

1. The scope of inspections performed on each SG,
2. Active degradation mechanisms found,
3. Nondestructive examination techniques utilized for each degradation mechanism,
4. Location, orientation (if linear), and measured sizes (if available) of service induced indications,
5. Number of tubes plugged during the inspection outage for each active degradation mechanism,
6. Total number and percentage of tubes plugged to date, and
7. The results of condition monitoring, including the results of tube pulls and in-situ testing.

b. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the NRC prior to returning the steam generators to service should any of the following 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, increased by estimated leakage by all other sources (alternate repair criteria and non-alternate repair criteria indications), exceeds the leak limit determined from the licensing basis dose calculation for the postulated main steamline break for the next operating cycle.
2. If ODSCC indications are identified that extend beyond the confines of the tube support plate.
3. 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.

(continued)

5.6 Reporting Requirements

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

c. The results of the inspection of W* tubes shall be reported to the Commission pursuant to 10 CFR 50.4 within 90 days following return to service of the steam generators. This report shall include:

   1. Identification of W* tube indications and indications that do not meet W* requirements and were plugged or repaired, including the following information: the number of indications, the locations of the indications (relative to the BWT and TTS), the orientation (axial, circumferential, volumetric, inclined), the radial position of the tube within the tubesheet, the W* Zone of the tube, the severity of each indication (estimated depth), the side of the tube in which the indication initiated (inside or outside diameter), the W* inspection distance measured with respect to the BWT or TTS (whichever is lower), the length of axial indications, the angle of inclination of clearly skewed axial cracks (if applicable), verification that the upper crack tip of W* indications returned to service in the prior cycle remain below the BWT by at least the 95% confidence NDE uncertainty on locating the crack tip relative to the BWT, updated 95% growth rate for use in operational assessment, the cumulative number of indications detected in the tubesheet region as a function of elevation within the tubesheet, and the condition monitoring and operational assessment main steamline break leak rate for each indication and each SG in accordance with the leak rate methodology described in PG&E Letter DCL-05-018, dated March 11, 2005, as supplemented by PG&E Letter DCL-05-090, dated August 25, 2005.

   2. Assessment of whether the results were consistent with expectations and, if not consistent, a description of the proposed corrective action.

d. The aggregate calculated steam line break leakage from application of all alternate repair criteria and non-alternate repair criteria shall be reported to the Commission pursuant to 10 CFR 50.4 within 90 days following return to service of the steam generators.

e. For implementation of the repair criteria for axial PWSCC at dented TSPs, the NRC shall be notified prior to startup, pursuant to 10CFR50.72, of the following conditions that indicate a failure of performance criteria:

   1. The calculated SG probability of burst for condition monitoring exceeds 1×10^{-2}.

   2. The calculated SG leakage for condition monitoring from all sources (all alternate repair criteria and non-alternate repair criteria indications) exceeds the leakage limit determined from the licensing basis steam line break dose calculation.

(continued)

5.6 Reporting Requirements

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

f. For implementation of the repair criteria for axial PWSCC at dented TSPs, the results of the condition monitoring and operational assessments will be reported to the NRC within 120 days following completion of the inspection. The report will include:

1. Tabulations of indications found in the inspection, tubes repaired, and tubes left in service under the ARC.
2. Growth rate distributions for indications found in the inspection and growth rate distributions used to establish the tube repair limits.
3. Plus Point confirmation rates for bobbin detected indications when bobbin is relied upon for detection of axial PWSCC in less than or equal to 2 volt dents.
4. For condition monitoring, an evaluation of any indications that satisfy burst margin requirements based on the Westinghouse burst pressure model, but do not satisfy burst margin requirements based on the combined ANL ligament tearing and throughwall burst pressure model.

(continued)

5.6 Reporting Requirements

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

5. Performance evaluation of the operational assessment methodology for predicting flaw distributions as a function of flaw size.
6. Evaluation results of number and size of previously reported versus new PWSCC indications found in the inspection, and the potential need to account for new indications in the operational assessment burst evaluation.
7. Identification of mixed mode (axial PWSCC and circumferential) indications found in the inspection and an evaluation of the mixed mode indications for potential impact on the axial indication burst pressures or leakage.
8. Any corrective actions found necessary in the event that condition monitoring requirements are not met.

g. For implementation of the probability of prior cycle detection (POPCD) method, for the voltage-based repair criteria at tube support plate intersections, if the end-of-cycle conditional main steamline break burst probability, the projected main steamline break leak rate, or the number of indications are underpredicted by the previous cycle operational assessment, the following shall be reported to the Commission pursuant to 10 CFR 50.4 within 90 days following return to service of the steam generators:

1. The assessment of the probable causes for the underpredications, proposed corrective actions, and any recommended changes to probability of detection or growth methodology indicated by potential methods assessments.
2. An assessment of the potential need to revise the alternate repair criteria analysis methods if: the burst probability is underpredicted by more than 0.001 (i.e., 10% of the reporting threshold) or an order of magnitude; or the leak rate is underpredicted by more than 0.5 gpm or an order of magnitude.
3. An assessment of the potential need to increase the number of predicted low voltage indications at the beginning of cycle if the total number of as-found indications in any SG are underestimated by greater than 15% or by greater than 150 indications.

Changes to Technical Specification Bases Pages

BASES

APPLICABLE SAFETY ANALYSES (continued)	RCS flow, RCP rotor seizure, and RCP shaft break events. For each of these events, it is demonstrated that all the applicable safety criteria are satisfied. For the remaining accident/safety analyses, operation of all four RCS loops during the transient up to the time of reactor trip is assured thereby ensuring that all the applicable acceptance criteria are satisfied. Those transients analyzed beyond the time of reactor trip were examined assuming that a loss of offsite power occurs which results in the RCPs coasting down. The plant is designed to operate with all RCS loops in operation to maintain DNBR above the Safety Limit value during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant. RCS Loops - MODES 1 and 2 satisfy Criterion 2 of 10 CFR 50.36 (c)(2)(ii).
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 at rated power. An OPERABLE RCS loop consists of one OPERABLE RCP for heat transport and the associated SG, OPERABLE ~~in accordance with the Steam Generator Tube Surveillance Program,~~ with a water level within the limits specified in SR 3.4.5.2, except for operational transients. A RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.
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.

(continued)

BASES (continued)

LCO

The purpose of this LCO is to require that at least two RCS loops be OPERABLE. In MODE 3 with the Rod Control System capable of rod withdrawal, two RCS loops must be in operation. Two RCS loops are required to be in operation in MODE 3 with the Rod Control System capable of rod withdrawal due to the postulation of a power excursion because of an inadvertent control rod withdrawal. The required number of RCS loops in operation ensures that the Safety Limit criteria will be met for all of the postulated accidents.

When the Rod Control System is not capable of rod withdrawal, only one RCS loop in operation is necessary to ensure removal of decay heat from the core and homogenous boron concentration throughout the RCS. An additional RCS loop is required to be OPERABLE to ensure that redundancy for heat removal is maintained.

The Note permits all RCPs to be removed from operation for ≤ 1 hour per 8 hour period. The purpose of the Note is to perform tests that are required to be performed without flow or pump noise. One of these tests 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 cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1, thereby maintaining the margin to criticality. Boron reduction 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.

(continued)

BASES

LCO (continued)	temperature difference limits the available relative energy source and the pressurizer level condition provides an expansion volume to accommodate possible reactor coolant thermal swell. These conditions are intended to prevent a low temperature overpressure event due to a thermal transient when a RCP is started. 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. A RHR loop is in operation when the pump is operating and providing forced flow through the loop. Because a loop can be operating without being OPERABLE, the LCO requires at least one loop OPERABLE and in operation.
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	<u>A.1 and A.2</u> If one required RCS loop is inoperable and two RHR loops are inoperable, redundancy for heat removal is lost. Action must be initiated to restore a second RCS loop or RHR loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

(continued)

LCO (continued)

The Note specifies that a RCP may be started if the pressurizer water level is less than 50%. This option of RCP start with pressurizer water level less than 50% supports plant operational flexibility. The open volume in the pressurizer provides space to sustain reactor coolant thermal swell without incurring a possible excessive pressure transient due to energy additions from the S/G secondary water. The purpose of conditions to allow initial RCP start when none is running is to prevent a possible low temperature RCS overpressure event due to a thermal transient when a RCP is started. The condition of SG/RCS temperature difference limits the available relative energy source and the pressurizer level condition provides an expansion volume to accommodate possible reactor coolant thermal swell. These conditions are intended to prevent a low temperature overpressure event due to a thermal transient when a 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~~.

APPLICABILITY

"Loops filled" is a condition in which natural circulation can be used as a backup means of decay heat removal if forced circulation via RHR is lost. RCS loops are considered filled when the RCS is capable of being pressurized to at least 150 psig, no gas has been directly injected into the RCS, and the RCS has not been drained below 112 ft (Ref. 2). In addition to these requirements, crediting heat removal via natural circulation requires at least two steam generators filled to ≥15% narrow range level and vented, or capable of being vented, to the atmosphere, and auxiliary feedwater available to add water to the relied-upon steam generators (Ref. 1). A loops filled condition is established at the completion of steam generator U-tube vacuum refill or after "bumping" RCPs.

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least two SGs is required to be ≥ 15%.

(continued)

There are no changes to this page. Page included for information only.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

Possible leakage from a Control Rod Drive Mechanism (CRDM) canopy seal weld may be construed as either identified or unidentified LEAKAGE but not construed as pressure boundary LEAKAGE in accordance with Westinghouse letter PGE-88-622.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leak tight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

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).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

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~~.

Insert 1

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 SGTR (Ref. 3) is more limiting for radiological releases at the site boundary. The radiological dose analysis assumes loss of off-site power at the time of reactor trip with no subsequent condenser cooling available. The steam generator (SG) PORV ~~for the SG~~ that has sustained the tube rupture is assumed to ~~be~~ open for 30 minutes, at which time ~~the RCS pressure is below the lift setting of~~ the PORV. The ~~dose~~ consequences resulting from the SGTR accident are within the limits defined in 10 CFR 100 (Ref. 6).

fail

operator closes the block valve to

Insert 2

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 10 CFR 50.36 (c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. <u>Pressure Boundary LEAKAGE</u>

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, gaskets, or the CRDM canopy seal welds is not pressure boundary LEAKAGE. Pressure boundary leakage is defined as "non-isolable" leakage. A "non-isolable" RCS leak is one that is not capable of being isolated from the RCS using installed automatic or accessible manual valves.

(continued)

BASES

LCO (continued)

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 monitoring equipment 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). Identified LEAKAGE does not include LEAKAGE from portions of the Chemical and Volume Control System outside of containment that can be isolated from the RCS. LEAKAGE of this nature may be reviewed for possible impact on the Primary Coolant Sources Outside Containment program. Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through All Steam Generators (SGs)

Total primary to secondary LEAKAGE amounting to 1 gpm through all SGs produces acceptable offsite doses in the SLB 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.

d

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

Insert 3

The 150 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.

The primary-to-secondary operational leakage limit of 150 gallons per day per steam generator is more restrictive than the standard operating leakage limits and is intended to provide an additional margin to accommodate a crack which might grow at a greater than expected rate or unexpectedly extent outside the thickness of the tube support plate. Hence, the reduced leakage limit, when

(continued)

BASES

LCO

e. <u>Primary to Secondary LEAKAGE through Any One SG</u> (continued)

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 shut down in a timely manner.

Calculations for primary-to-secondary leakage are performed using approximate Standard Reference State of 25°C. When determining primary-to-secondary leakage of 150 gallons per day, indeterminant inaccuracies associated with determination of leakage are not considered.

d

For MODES 3 and 4, the primary system radioactivity level (source term) may be very low, making it difficult to measure primary-to-secondary leakage of 150 gallons per day. Therefore, if steam generator water samples indicate less than the minimum detectable activity of $5.0*E^{-7}$ microcuries/ml for each principal gamma emitter, the leakage requirement of Specification 3.4.13 e may be considered met.

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.

d

A note has been added to the APPLICABILITY. For MODES 3 and 4, the primary system radioactivity level (source term) may be very low, making it difficult to measure primary-to-secondary leakage of 150 gallons per day. Therefore, if steam generator water samples indicate less than the minimum detectable activity of $5.0*E^{-7}$ microcuries/ml for each principal gamma emitter, the leakage requirement of Specification 3.4.13 e. may be considered met.

ACTIONS

<u>A.1</u>

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

(continued)

or primary to secondary LEAKAGE is not within limit,

BASES

ACTIONS

A.1 (continued)

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

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.

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. 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.

1 states

The Surveillance is modified by two Notes.

The RCS water inventory balance must be met with the reactor at steady state operating conditions (stable temperature, power level, pressurizer level, 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. The 12 hour allowance provides sufficient time to collect and process all 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 (continued)

operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature (Tavg changes less than 5°F per hour) power level, pressurizer and makeup tank levels, makeup and letdown (balanced with no diversion to LHUTS), 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 by the containment structure sump level and flow monitoring system. It should be noted that LEAKAGE past seals, gaskets or CRDM canopy seal welds is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

Insert 4

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The 12 hour Frequency after steady state operation has been achieved provides for those situations following a transient such that the 72 hours plus extension allowed by SR 3.0.2 would be exceeded. Under these circumstances, the SR would be due within 12 hours after steady state operation has been reestablished and every 72 hours thereafter during steady state operation.

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.

SR 3.4.13.3 2

Insert 5

This SR provides a requirement to determine primary-to-secondary leakage once every 72 hours while operating in MODES 1, 2, 3, or 4. During normal operation this will be done using a correlation of radioactivity at the steam jet air ejectors. During periods of significant source term and mass flow rate changes or when the primary system radioactivity levels are low engineering judgment may be used to aid in determining leakage rates. The 72 hour frequency is adequate to allow early detection of a significant primary-to-secondary leakage and allow the plant to shutdown in a timely manner reducing the risk of a tube rupture.

(continued)

BASES (continued)

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.
5. NUREG-1061, Volume 3, November, 1984.
6. 10 CFR 100.

Insert 6

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.16.2

This Surveillance is modified by a Note. The Note modifies the surveillance to allow entry into and operation in MODE 3 ≥ 500°F and MODE 2 prior to performing this Surveillance Requirement.

This Surveillance is performed to ensure iodine remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering gross activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide less indicative results.

SR 3.4.16.3

A radiochemical analysis for $\bar{E}$ determination is required every 184 days (6 months) with the plant operating in MODE 1 equilibrium (as defined in SR 3.4.16.3 NOTE) conditions. The $\bar{E}$ determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for $\bar{E}$ is the qualitative measurement of the specific activity for each radionuclide, except for radionuclides with half-lives less than 10 minutes and all radioiodines which are identified in the reactor coolant. The specific activity for these individual radionuclides shall be used in the determination of $\bar{E}$ for the reactor coolant sample. Determination of the contributors to $\bar{E}$ shall be based upon those energy peaks identifiable with a 95% confidence level. The Frequency of 184 days recognizes $\bar{E}$ does not change rapidly.

This SR has been modified by a Note that indicates sampling for $\bar{E}$ determination is required to be performed within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures that the radioactive materials are at equilibrium so the analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar abnormal event.

REFERENCES

1. 10CFR100.11, 1973.
2. FSAR, Sections 15.4.3 and 15.5.20.

Insert 7 (for new B 3.4.17) →

Technical Specification Bases Inserts

Insert 1

that prior to the accident there is 1 gpm total primary to secondary LEAKAGE to all steam generators (SGs) as an initial condition. 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.

Insert 2

The safety analyses for the SLB accident (Ref. 3) assumes that prior to the accident there is 1 gpm total primary to secondary LEAKAGE to all SGs as an initial condition, which bounds the allowable operational LEAKAGE rate limits in LCO 3.4.13.d. The dose consequences resulting from the SLB accident are within a small fraction of the limits defined by 10 CFR 100.

Insert 3

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.

Insert 4

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.

Insert 5

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, compliance with LCO 3.4.17, "Steam Generator Tube Integrity," should be evaluated. 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).

<u>Insert 6</u>

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

8. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

<u>Insert 7</u>

See next page for new TS 3.4.17 Bases.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.17 Steam Generator (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

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 that prior to the accident there is 1 gpm total primary to secondary LEAKAGE to all SGs as an initial condition, which bounds the allowable operational LEAKAGE rate limits in LCO 3.4.13, "RCS Operational LEAKAGE". The analysis assumes following the accident, there is 150 gpd leakage to each of the three intact SGs, and there is accident induced leakage to the ruptured SG associated with a double-ended rupture of a single tube. The SGTR radiological dose analysis assumes loss of off-site power at the time of reactor trip with no subsequent condenser cooling available. The SG PORV for the SG that has sustained the tube rupture is assumed to fail open for 30 minutes, at which time the operator closes the block valve to the PORV. The SGTR radiological dose analysis assumes the contaminated secondary fluid is released briefly to the atmosphere from all the PORVs following reactor trip, is released from the ruptured SG PORV for 30 minutes, is released from the intact SG PORVs during the cooldown, and is released from all PORVs following cooldown until termination of the event.

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 an assumed 1 gpm total primary to secondary LEAKAGE to all SGs as an initial condition prior to the accident. 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 in accordance with the Steam Generator Program.

(continued)

BASES

LCO (continued)

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 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.

(continued)

LCO (continued)

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 main steam line break accident analysis assumes that accident induced leakage following the accident does not exceed 10.5 gpm to the faulted SG, except for specific types of degradation at specific locations as described in paragraph 5.5.9.c of the Steam Generator Program in which leakage from application of all alternate tube repair criteria degradation and non-alternate tube repair criteria degradation is not to exceed 10.5 gpm. To support the accident induced leakage performance criteria, the non-alternate repair criteria degradation accident induced leakage following the accident in the faulted SG is conservatively limited to 1 gpm at hot conditions.

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.

(continued)

ACTIONS

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

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program as required by SR 3.4.17.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, 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 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.

(continued)

ACTIONS (continued)

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.

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, nondestructive 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

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

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 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."