50-275/323

February 19, 199

Mr. Gregory M. Rueger
Senior Vice President and General Manager
Pacific Gas and Electric Company
Diablo Canyon Nuclear Power Plant
P. O. Box 3
Avila Beach, California 93424

SUBJECT: ISSUANCE OF AMENDMENTS FOR DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 1 (TAC NO. M98283) AND UNIT NO. 2 (TAC NO. M98284)

Dear Mr. Rueger:

The Commission has issued the enclosed Amendment No. 129 to Facility Operating License No. DPR-80 and Amendment No. 127 to Facility Operating License No. DPR-82 for the Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated March 10, 1997, as supplemented by letters dated May 20, 1997; March 13, August 28, and October 22, 1998 and January 29, and February 2, 1999.

These amendments revise TS 3/4.4.5 and its associated Bases to allow the implementation of steam generator (SG) tube alternate repair criteria for axial indications in the Westinghouse explosive tube expansion (WEXTEX) region below the top of the tubesheet and below the bottom of the WEXTEX transition that may exceed the current TS depth-based plugging limit.

A copy of the related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,
Original Signed By
Steven D. Bloom, Project Manager
Project Directorate IV-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-275
and 50-323

Enclosures: 1. Amendment No. 129 to DPR-80
2. Amendment No. 127 to DPR-82
3. Safety Evaluation

cc w/encls: See next page

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DOCUMENT NAME: DC98283.WPD

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-275

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 129
License No. DPR-80

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Pacific Gas and Electric Company (the licensee) dated March 10, 1997, as supplemented by letters dated May 20, 1997, March 13, August 28, and October 22, 1998, and January 29 and February 2, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-80 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 129 , are hereby incorporated in the license. Pacific Gas and Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of its date of issuance and is to be implemented within 30 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven D. Bloom, Project Manager
Project Directorate IV-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 19, 1999

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-323

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 127
License No. DPR-82

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Pacific Gas and Electric Company (the licensee) dated March 10, 1997, as supplemented by letters dated May 20, 1997, March 13, August 28, and October 22, 1998, and January 29 and February 2, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-82 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 127 , are hereby incorporated in the license. Pacific Gas and Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of its date of issuance and is to be implemented within 30 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven D. Bloom, Project Manager
Project Directorate IV-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 19, 1999

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 129 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 127 TO FACILITY OPERATING LICENSE NO. DPR-82

DOCKET NOS. 50-275 AND 50-323

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by Amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
3/4 4-11	3/4 4-11
3/4 4-12	3/4 4-12
3/4 4-14	3/4 4-14
3/4 4-14b	3/4 4-14b
---	3/4 4-14c
---	3/4 4-14d
3/4 4-15	3/4 4-15
3/4 4-15a	3/4 4-15a
3/4 4-16*	3/4 4-16*
B 3/4 4-3	B 3/4 4-3
B 3/4 4-3a	B 3/4 4-3a
---	B 3/4 4-3b
B 3/4 4-4	B 3/4 4-4

*No changes were made to this page. Reissued to become a one-sided page.

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.*#

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirement of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

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

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

   2) All nonplugged tubes that previously had detectable wall penetrations (greater than 20%).

* - Amendment Nos. 129 and 127 applicable for Units 1 and 2, Cycles 10 and 11 only.

- In-situ testing will be performed in accordance with PG&E letter DCL-98-148 dated October 22, 1998.

2) Tubes in those areas where experience has indicated potential problems.

3) A tube inspection (pursuant to Specification 4.4.5.4a.8) 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.

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.

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

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1) 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

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

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 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;

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall 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 4.4.5.3a. The interval may then be extended to a maximum of once per 40 months; and

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

1) Reactor-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2; or

2) A seismic occurrence greater than the Double Design Earthquake, or

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

4) A main steam line or feedwater line break.

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1) 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;

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

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

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

5) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;

6) 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.

a) This definition does not apply to tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to Specification 4.4.5.4a.10) for the repair limit applicable to these intersections.

b) 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 Specification 4.4.5.4a.11).

7) 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 4.4.5.3c., above;

8) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg;

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

10) 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:

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 4.4.5.4a.10)c below.

c. Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit [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 Westinghouse letter to PG&E dated September 3, 1992, "Deformation of Steam Generator Tubes Following a Postulated LOCA and SSE Event," 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 4.4.5.4a.10)a, 4.4.5.4a.10)b, and 4.4.5.4a.10)c. The mid-cycle repair limits are determined from the following equations:

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

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

where:

V_{URL}	= upper voltage repair limit
V_{LRL}	= lower voltage repair limit
V_{MURL}	= mid-cycle upper voltage 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)
VSL	= 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 4.4.5.4a.10)a, 4.4.5.4a.10)b, and 4.4.5.4a.10)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.

11) 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:

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 to the tubesheet below the BWT that precludes tube pull out in the event of a 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, Revision 1, 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 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 equal to or greater than 40% degradation within or below the W* length that is left in service, and degraded within the limits specified in Specification 4.4.5.4a.11)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 below the TTS 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 tubesheet secondary face.

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 tubve 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 RPC C-scan, such that an angular measurement is required, and the measured angle exceeds the measurement uncertainty of ΔNDE_{CA}.

SURVEILLANCE REQUIREMENTS (Continued)

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 of combination of the tube degradation below the W* length is acceptable.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit) required by Table 4.4-2.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following completion of the inspection. This Special Report shall include:

   1) Number and extent of tubes inspected,

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

   3) Identification of tubes plugged.

c. Results of steam generator tube inspections, which fall into Category C-3, shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

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 conditions arise:

   1) If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution, reduced by estimated leakage by all other alternate repair criteria, exceeds the leak limit determined from the licensing basis dose calculation for the postulated main steamline break for the next operating cycle.

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

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

e. The results of the inspection of W* tubes shall be reported to the Commission pursuant to Specification 6.9.2 within 90 days following return to service of the steam generators. This report shall include:

1) Identification of W* tubes.

2) W* inspection distance measured with respect to the BWT or the top of the tubesheet, whichever is lower.

3) Elevation and length of axial indications within the flexible W* distance and the angle of inclination of clearly skewed axial cracks (if applicable).

4) The total steam line break leakage for the limiting steam generator per WCAP-14797.

f. The aggregate calculated steam line break leakage from application of all alternate repair criteria shall be reported to the Commission pursuant to Specification 6.9.2 within 90 days following return to service of the steam generators.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS 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 steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2. The other steam generator 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 steam generators 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.

STEAM GENERATORS (Continued)

mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (primary-to-secondary leakage = 150 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. DCPP has demonstrated that primary-to-secondary leakage of 150 gallons per day per steam generator can readily be detected during power operation. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Operating experience of tubes with through-wall cracking in the transition region of European plants and in-situ pressure testing of a domestic plant with primary water stress corrosion cracking in the roll transitions suggests that leakage at operating conditions from W* tubes would not be expected.

The W* criteria incorporate the guidance provided in Revision 1 of WCAP-14797, "Generic W* Tube Plugging Criteria for 51 Series Steam Generator Tubesheet Region WEXTEX Expansions." W* length is the distance into the tubesheet below the bottom of the WEXTEX transition (BWT) that precludes tube pullout in the event of a complete circumferential separation of the tube below the W* length.

For tubes to which the W* criteria are applied, indications of degradation in excess of 40% through-wall can remain in service without a loss of functionality or structural and leakage integrity. Tubes to which W* is applied can experience through-wall degradation up to the limits defined in Revision 1 of WCAP-14797 without increasing the probability of a tube rupture or large leakage event. The guidance of Regulatory Guide 1.121, issued for coment in August 1976, is used to assess the limits of acceptable tube degradation within W*. A potential exists for W* tubes to allow primary-to-secondary leakage during a postulated steam line break. Information is provided in Revision 1 of WCAP-14797 that is used to calculate the expected leakage at steam line break conditions for W* tubes. Tube degradation of any extent below the W* length, including a complete circumferential separation of the tube, is acceptable and does not require repair.

Axial cracks in tubes returned to service using the W* criteria must remain below the secondary tubesheet face at the end of the subsequent operating cycle. This performance criteria is demonstrated by operational assessment and condition monitoring.

STEAM GENERATORS (Continued)

The combined calculated leak rate from all alternate repair criteria must be less than the maximum allowable SLB leak rate limit in any one steam generator in order to maintain off-site doses to within 10 CFR 100 guideline values during a postulated steam line break event.

The voltage-based repair limits of SR 4.4.5.4a.10) implement the guidance in GL 95-05 and are applicable only to Westinghouse-designed steam generators (SGs) with outside diameter stress corrosion cracking (ODSCC) located at the tube-to-tube support plate intersections. The voltage-based repair limits are not applicable to other forms of SG tube degradation nor are they applicable to ODSCC that occurs at other locations within the SG. Additionally, the repair criteria apply only to indications where the degradation mechanism is dominantly axial ODSCC with no significant cracks extending outside the thickness of the support plate. Refer to GL 95-05 for additional description of the degradation morphology.

Implementation of SR 4.4.5.4a.10) requires a derivation of the voltage structural limit from the burst versus voltage empirical correlation and then the subsequent derivation of the voltage repair limit from the structural limit (which is then implemented by this surveillance).

The voltage structural limit is the voltage from the burst pressure/bobbin voltage correlation, at the 95-percent prediction interval curve reduced to account for the lower 95/95-percent tolerance bound for tubing material properties at 650°F (i.e., the 95-percent LTL curve). The voltage structural limit must be adjusted downward to account for potential flaw growth during an operating interval and to account for NDE uncertainty. The upper voltage repair limit, V_{URL}, is determined from the structural voltage limit by applying the following equation:

$$V_{URL} = V_{SL} - V_{Gr} - V_{NDE}$$

where V_{Gr} represents the allowance for flaw growth between inspections and V_{NDE} represents the allowance for potential sources of error in the measurement of the bobbin coil voltage. Further discussion of the assumptions necessary to determine the voltage repair limit are discussed in GL 95-05.

The mid-cycle equation in SR 4.4.5.4a.10)e. should only be used during unplanned inspections in which eddy current data is acquired for indications at the tube support plates.

SR 4.4.5.5d implements several reporting requirements recommended by GL 95-05 for situations which the NRC wants to be notified prior to returning the SGs to service. For the purposes of this reporting requirement, leakage and conditional burst probability can be calculated based on the as-found voltage distribution rather than the projected end-of-cycle voltage distribution (refer to GL 95-05 for more information) when it is not practical to complete these calculations using the projected EOC voltage distributions prior to returning the SGs to service. Note that if leakage and conditional burst probability were calculated using the measured EOC voltage distribution for the purposes of addressing the GL Section 6.a.1 and 6.a.3 reporting criteria, then the results of the projected EOC voltage distribution should be provided per the GL Section 6.b(c) criteria.

STEAM GENERATORS (Continued)

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a wastage defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit defined in Surveillance Requirement 4.4.5.4a. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission as a Special Report pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are functionally consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The primary-to-secondary operational leakage limit of 150 gpd 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 extend outside the thickness of the tube support plate. Hence, the reduced leakage limit, when combined with an effective leak rate monitoring program, provides additional assurance that should a significant leak be experienced in service, it will be detected, and the plant 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 gpd, indeterminent inaccuracies associated with determination of leakage are not considered.

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 principal gamma emitters, the 150 gallons per day leakage limit may be considered met.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 GPM with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the safety analyses.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 129 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 127 TO FACILITY OPERATING LICENSE NO. DPR-82

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR POWER PLANT, UNITS 1 AND 2

DOCKET NOS. 50-275 AND 50-323

1.0 INTRODUCTION

By application dated March 10, 1997, as supplemented by letters dated May 20, 1997, March 13, August 28, and October 22, 1998, and January 29, and February 2, 1999, Pacific Gas and Electric Company (the licensee) submitted a proposed license amendment to implement alternate steam generator tube repair criteria ("W star" (W*)) at Diablo Canyon Power Plant, Units 1 and 2. The license amendment consists of modifications to the technical specifications (TS) to permit tubes with defects located in tubesheet expansions to remain in service provided certain conditions are satisfied. To support its amendment, the licensee submitted WCAP-14797, Revision 1, "Generic W* Tube Plugging Criteria for 51 Series Steam Generator Tubesheet Region WEXTEX Expansions."

The March 13 and August 28, 1997, and October 22, 1998, and January 29 and February 2, 1999, supplemental letters provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination published in the Federal Register on November 19, 1997 (62 FR 61843).

2.0 BACKGROUND

2.1 Regulatory Framework For Proposed Licensing Action

Steam generator tubing comprises a significant fraction of the reactor coolant pressure boundary. Criterion 14 of the General Design Criteria (GDC) in Appendix A to Part 50 of Title 10 of the Code of Federal Regulations (CFR) states that the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. Section 50.55a(c) specifies that components that are part of the reactor coolant pressure boundary must be designed and constructed to meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. To ensure the continued integrity of the tubing at operating pressure water reactor (PWR) facilities, 10 CFR 50.55a further requires that throughout the service life of a PWR facility,

Class 1 components meet the requirements in Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" of the ASME Code. This requirement includes the inspection and tube repair criteria of Section XI of the ASME Code. However, an exception is provided for design and access provisions and preservice examination requirements in Section XI. In addition, 10 CFR 50.55a(b)(2)(iii) states that if the technical specification surveillance requirements for steam generators differ from those in Article IWB-2000 of Section XI of the ASME Code, the inservice inspection program is governed by the technical specifications.

As part of the plant licensing basis, PWR applicants analyze the consequences of postulated design basis accidents that assume degradation of the steam generator tubes such as primary coolant leaks to the secondary coolant side of the steam generators. Examples of such accidents are a steam generator tube rupture, a main steam line break, a locked rotor, and a control rod ejection. Analyses of these accidents consider the primary-to-secondary leakage that may occur during these postulated events when demonstrating that radiological consequences do not exceed the 10 CFR Part 100 guidelines, or some fraction thereof, for offsite doses, nor GDC-19 for control room operator doses. The staff uses criteria specified in NUREG-0800, the Standard Review Plan, to evaluate these accidents.

A plant's technical specifications require that licensees perform periodic inservice inspections of the steam generator tubing and repair or remove from service (by installing plugs in the tube ends) all tubes exceeding the tube repair limit. In addition, operational leakage limits are included in the technical specifications to ensure that, should tube leakage develop, the licensee will take prompt action to avoid rupture of the leaking tubes. These requirements are intended to ensure that burst margins are maintained consistent with Appendices A and B to 10 CFR Part 50 and that the potential for leakage is maintained consistent with what has been analyzed as part of the plant licensing basis.

Revision 1 of NRC Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," provides guidance concerning steam generator tube inspection scope and frequency and nondestructive examination (NDE) methodology. Regulatory Guide 1.83 is referenced in the standard review plan and is intended to provide a basis for reviewing inservice inspection criteria in the technical specifications. NRC Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," provides guidelines for determining the tube repair criteria and operational leakage limits that are specified in the technical specifications. Together, these two regulatory guides provide specific criteria that should be considered in proposed steam generator tube alternate repair criteria in order to satisfy the previously mentioned regulatory requirements.

2.2 Description of Proposed Repair Criteria

The steam generators at Diablo Canyon Power Plant are Westinghouse Model 51 with a U-tube configuration. Each tube is secured in the tubesheet above the lower plenum of the steam generator by an explosive expansion process (WEXTEX). The WEXTEX process expands each tube over its entire length of the tubesheet and forms an interference fit between the tube and tubesheet. This interference fit forms the interface which provides the structural and part of the leaktight boundary between the primary and secondary systems at each end of a steam generator tube. Located near the top of the tubesheet is a region where the tube transitions from the tubesheet hole diameter to that of the original tube (e.g., ⅞-inch outside diameter).

The tube material in the WEXTEX expanded region has high residual stresses which contribute to the material's susceptibility to stress corrosion cracking. Experience has shown that the state of stress in WEXTEX expansion transitions in Alloy 600 steam generator tubing creates an environment favorable to the initiation and growth of primary water stress corrosion cracking (PWSCC).

The proposed W* repair criteria would permit some tubes with predominantly axially oriented PWSCC in WEXTEX tubesheet expansions to remain in service. The repair criteria include restrictions on the type and location of flaws that minimize the likelihood of leakage or structural failure under normal operating or postulated accident conditions. Specifically, tubes that are returned to service using W* (W* tubes) should be limited to those tubes with flaws located below the bottom of the WEXTEX transition (BWT). The BWT is considered to be the highest elevation in the tubesheet where the tube and tubesheet are in contact. Allowances are also included to account for NDE uncertainty and crack growth over the next cycle of operation. Cracks must be primarily oriented in the axial direction, and the repair criteria do not apply to volumetric degradation (e.g., intergranular attack). Structural integrity is maintained by ensuring each W* tube has sufficient length of undegraded expansion below the tubesheet expansion transition. Cracking may be present at different axial locations in the tubesheet expansion, but degraded areas are assumed to have no contribution toward preventing the tube from pulling out of the tubesheet due to tube endcap loadings. The restriction to limit the proposed repair criteria to axial cracking minimizes the potential for an axial tube separation at a flaw within the tubesheet. The existence of the tubesheet surrounding tube flaws precludes structural failure by burst due to primary-to-secondary pressure.

The extent of degradation within the WEXTEX may vary from tube to tube. Consequently, the licensee has proposed to inspect tubes as potential candidates for W* over varying lengths ("flexible W* length") in order to assess their condition. The flexible W* length represents the sum of the undegraded length of tube necessary to ensure structural integrity (W* length) plus allowances for NDE uncertainties and crack growth. Different W* lengths apply to the hot leg and cold leg sides of the steam generator and to defined regions of the tubesheet. Multiple W* lengths are proposed to account for varying thermal and mechanical conditions that may exist during postulated accident conditions. All indications left in service within the flexible W* length must be inspected with a rotating probe during all future refueling outages.

Although WEXTEX expansions are quite resistant to any axial movement of a steam generator tube relative to the tubesheet, their ability to provide a pressure retaining seal to form a leak tight seal is limited. The degree of interference between a steam generator tube and the tubesheet may not prevent primary-to-secondary leakage under all conditions. As such, the proposed repair criteria includes a proposed methodology to quantify the total leakage of all flaws returned to service using W*. Flaws located near the tubesheet secondary face will provide the greatest contribution to the calculated leak rate. All flaws are assumed to be through-wall for purposes of determining the expected leak rate under accident conditions.

Following each inservice inspection, the licensee will complete an assessment to estimate the projected end-of-cycle leakage from tubes returned to service using the proposed repair criteria. The leak rate calculation may be performed using either deterministic or probabilistic methodologies. The deterministic approach incorporates conservative margins in the results by

assuming empirical factors relevant to the model at 95 percent prediction intervals. The Monte Carlo method calculates an upper 95 percent bound on the total leak rate from W* tubes. The deterministic calculations should yield a more conservative prediction of end-of-cycle leakage than that from a Monte Carlo analysis. The calculated leakage from W* tubes combined with other sources of postulated leakage (e.g., cracking at tube support plate intersections) from other tubes returned to service by approved repair criteria must remain below the plant specific allowable leakage limit. If the total leakage is less than the allowable limit, then tubes with indications may be returned to service provided that all other W* criteria are satisfied. If the allowable leakage limit is exceeded, tubes shall be plugged until the allowable leakage limit is met. As documented in Amendment Nos. 124 and 122 to Operating License No. DPR-80 and Operating License No. DPR-82, respectively, issued March 12, 1998, the allowable leakage limit under steam line break conditions for the Diablo Canyon units is 12.8 gallons per minute. In the required 90 day outage report, the licensee shall report to the NRC the total leakage for the limiting steam generator per WCAP-14797.

The model developed to estimate primary-to-secondary leakage is empirically based and represents a combination of two relationships. Industry experience with the ability of the model to accurately predict steam generator tube leak rates is unavailable at this time. In order to obtain the data to demonstrate the accuracy (or conservatism) in the leak rate model, the proposed amendment to implement the W* repair criteria is limited to two cycles of operation. During this interval, the licensee will attempt to obtain leak rate data through in-situ pressure testing of flaws in WEXTEX expansions. NDE criteria have been established to identify tubes as candidates for the in-situ pressure test program that have the potential for leakage during testing based on industry developed guidelines. The licensee may seek the permanent approval once the capabilities of the leak rate model are demonstrated.

The proposed repair criteria rely on results obtained from NDE methods to determine the mode of degradation (i.e., PWSCC), the location of cracks with respect to the BWT, and the orientation and length of identified flaws. The accurate determination of these factors will ensure that the assumptions of the repair criteria remain valid. The licensee's vendor, Westinghouse, completed a qualification program to quantify the uncertainty associated with performing the various measurements necessary for implementation of W*. Results were incorporated into the proposed repair criteria by applying an error determined at a one-sided 95% upper bound to the nominal value determined during the qualification program. W* inservice inspection results will be reported to the NRC in the required 90 day report.

3.0 EVALUATION

Steam generator tube repair limits within technical specifications are intended to ensure adequate structural and leakage integrity for a portion of the reactor coolant system. The depth-based repair limit is typically used for most modes of steam generator tube degradation. This limit establishes a conservative level at which tubes should be removed from service to minimize the potential for excessive primary-to-secondary leakage and gross rupture of a steam generator tube throughout the next cycle of operation. The application of depth-based repair limits in certain areas of a steam generator may lead to removing tubes from service that will continue to have adequate margins for structural and leakage integrity. Other factors such as the presence of a tubesheet or an inability for the degradation to progress further into a tube may enable consideration of alternate tube repair criteria. The W* repair criteria are intended to

address one such mode of degradation affecting the Diablo Canyon steam generator tubes. The location of the degradation (i.e, within the tubesheet) minimizes the potential for tube burst and may limit the primary-to-secondary leakage. The following summarizes the staff's evaluation of the proposed repair criteria.

3.1 Assessment of Structural Margins for W*Tubes

The proposal to permit tubes with confirmed steam generator tube degradation to remain in service must demonstrate that tubes returned to service using the repair criteria will maintain adequate structural margins. Regulatory Guide 1.121 includes specific loading criteria that licensees may adhere to in order to demonstrate the structural integrity of degraded steam generator tubing. Tube rupture and the pullout of a tube from the tubesheet are two potential modes of structural failure considered for tubes returned to service under the W* repair criteria.

Axial flaws permitted to remain in service in tubes returned to service using W* repair criteria would be located within the steam generator tubesheet at the beginning of the cycle. If the entire flaw remains below the tubesheet secondary face, the presence of the surrounding tubesheet will prevent tube rupture. In order to challenge the structural margins of a degraded W* tube, the upper tip of an inservice crack would have to propagate substantially in excess of the maximum observed growth rates determined from an analysis of previous inspection results. The establishment of a bounding flaw growth rate was determined using available data applicable to the mode of degradation found in WEXTEX expansions. The upper bound growth rate will be reassessed at each inspection to minimize the potential for crack growth into the freespan region above the tubesheet.

In the unlikely event that cracks grow into the freespan, the staff considered the conditions that would be necessary to structurally fail a W* tube. Spontaneous steam generator tube rupture is primarily a function of crack geometry (e.g., length), the differential pressure across the wall of the tube, and the crack's location. In order to maintain adequate margins for tube burst under all conditions, axial, through-wall, tube flaws must exceed a given length, typically on the order of one-half inch or larger, and have no external restraint (i.e., freespan). The W* repair criteria provides reasonable assurance that flaws will remain confined within the tubesheet during operation. However, in the event that high crack growth rates do develop, cracks would have to extend a significant distance above the tubesheet to measurably degrade the margins of structural integrity for the affected tube. Such crack growth rates are unlikely to occur. Therefore, flaws remaining in service by application of the W* repair criteria should have acceptable margins for tube burst.

The staff notes that other factors further reduce the potential for challenging steam generator tube structural margins from flaws returned to service under W*. Tube structural limits are determined assuming cracks penetrate through the entire wall of the tube. Therefore, the previous discussion assumes that the entire segment of the flaw that is located above the tubesheet is through-wall. Part through-wall flaws would require additional length in order to become susceptible to spontaneous failure based on empirical models for tube burst. Another consideration not credited in the staff's assessment is the reinforcing effect provided at the lower end of the freespan flaw by the tubesheet. Restricting the radial expansion at one end of a flaw would further elevate the expected burst pressure of a degraded tube.

The other postulated failure mode for W* tubes is pullout from the tubesheet under high axial tube loading. Differential pressures from the primary to the secondary sides of the steam generator impart axial loads into each tube that are reacted at the tubesheet interface. The peak postulated loading occurs during events involving a depressurization of the secondary side of the steam generator. However, normal operating loads can also be significant. The analysis supporting the licensee's proposed modification to the plant technical specifications addressed the limiting conditions necessary to maintain adequate structural integrity of the tube-to-tubesheet joint. Specifically, the tube must not experience excessive displacement relative to the tubesheet under bounding loading conditions with appropriate factors of safety considered.

The presence of circumferentially-oriented degradation within a steam generator tube under axial loading decreases the load bearing capability of the affected tube. If a tube becomes sufficiently degraded, these loads could lead to an axial separation of the tube. The W* repair criteria permit only axially-oriented degradation to remain in service provided it is located below the bottom of the WEXTEX transition or tubesheet secondary face, whichever is at a lower elevation. Axial tube loads are reacted by the tube over a span of tubing in the uppermost portion of the tubesheet. This structural reaction length is termed the "flexible W* length." Any circumferentially-oriented flaws in the flexible W* length of the tube would increase the potential for axial tube separation. Therefore, all tubes with circumferential and volumetric degradation within the W* inspection zone are excluded from the application of W*.

Axial flaws located within the W* length will permit axial tube loads to be transferred to undegraded portions of the tube located below the crack without increasing the potential for axial tube separation. The analytical assessment of W* loads conservatively assumed that the length of tube containing confirmed axial cracks does not contribute to the resistance of these loads. To account for this reduction in load bearing capability, the W* length is increased by the overall length of axial flaws within the W* region as well as allowances for NDE uncertainty and crack growth.

The licensee completed an assessment using analytical calculations and experimental results of tubes with degradation in the W* inspection zone to evaluate the structural margins of degraded W* tubes. Steam generator tubes were expanded through the WEXTEX expansion process in prototypical tubesheets. The test specimens were subjected to internal pressurization and axial loading at various temperature conditions in order to demonstrate acceptable structural capabilities under a range of loading conditions. Steam generator tubes secured into tubesheets through explosive expansion processes are not as rigidly fastened as are tubes that are mechanically expanded (i.e., hard rolled) as found in some steam generator designs. Upon increasing the temperature of the system and the internal pressure of the tube, the tube expands into the tubesheet creating a tighter interference fit between the tube and tubesheet. The licensee did not consider the reinforcing effect provided by the WEXTEX expansion. Rather, all contact forces between the tube and tubesheet are assumed to result from the thermal expansion differences between the tube and tubesheet and pressure acting on the inner surface of the tube. W* test specimens were constructed with conservative geometries with respect to the final proposed repair criteria. Despite using configurations with lower structural capabilities than expected of actual in-service steam generator tubes, the test program demonstrated that the specimens resisted pullout from the tubesheet in excess of the loading margins specified in Regulatory Guide 1.121.

The W* repair criteria was established, in part, on minimizing the potential for the growth of cracks into the freespan region above the tubesheet and maintaining adequate pullout strength from the tubesheet. The confinement of the surrounding tubesheet for all flaws left in service using this proposed alternate repair criteria will prevent tube structural failure by bursting. Limiting the repair criteria to only axially oriented flaws and inspecting over the entire flexible W* length should ensure that tube pullout from the tubesheet under limiting conditions is precluded with additional necessary margins of safety. On these bases, the staff has concluded that tubes returned to service using the W* repair criteria will maintain adequate structural integrity margins.

3.2 Steam Generator Tube Accident Leakage Integrity Assessment

The depth-based steam generator tube repair limits included in PWR technical specifications limit the depth of tube degradation to maintain, in part, a leak tight reactor coolant pressure boundary. The W* repair criteria is not intended to preclude primary-to-secondary leakage under all conditions. Rather, it will limit the leakage from all tubes returned to service to below plant specific allowable levels. The licensee will determine the potential end-of-cycle leakage under postulated main steam line break (MSLB) conditions for the next cycle of operation upon completion of each inservice inspection. The combined leakage from W* tubes and other tubes returned to service using other approved repair criteria should remain within acceptable levels. If the calculated end-of-cycle accident leakage is in excess of these levels, tubes shall be plugged until the allowable accident leakage limit is met. The licensee is required to submit to the NRC a summary of the leak rate estimate from the operational and condition monitoring assessments in accordance with TS 4.4.5.5.e and TS 4.4.5.5.f.

W* tube leakage is calculated using analytical models supported by data obtained from leak testing. The leakage model yields a calculated leak rate for a single, flawed steam generator tube that is the result of two models operating in series. These models separately determine the pressure drop for primary-to-secondary leakage through the crack and the tubesheet crevice. A computer code iteratively determines the crack exit pressure yielding the overall leak rate based on the crack inlet and tubesheet crevice outlet (i.e., secondary system) pressures. The first model estimates the leak rate through a crack in a WEXTEX expansion assuming the tube leaks into the freespan. This leak rate is modified to account for the confining effect from the tubesheet preventing the flaw or flaws from opening under high primary-to-secondary differential pressures.

The second part of the leakage model determines the pressure drop in the fluid as it passes from the exit of the crack through the crevice up to the tubesheet secondary face. This pressure loss is modeled using existing fluid mechanics analytical expressions. Empirical coefficients were determined by leak testing a number of specimens with varying tubesheet crevice lengths. As documented in WCAP 14797, Revision 1 (Proprietary), the correlation of the empirically derived data to develop these coefficients shows a relatively large degree of scatter. Standard normality tests have shown that the data can be represented as a normal distribution. However, some tests indicate otherwise. Because the significance of the departure from a normal distribution is minimal in alternative tests, it does not invalidate the assumption that the data are normally distributed. However, it does indicate that the normal approximation is weak.

Prior efforts to analytically calculate leak rates from stress corrosion cracking in piping and tubing have often been difficult due to the large range of variability exhibited by actual leak rate data. The conservatism of the leak rate model proposed to estimate leakage from W* tubes cannot be quantitatively assessed at this time due to a lack of leak rate data for cracking in WEXTEX tubesheet expansions. In order to acquire data necessary to evaluate the W* leak rate model, the licensee has proposed to limit the duration of the W* repair criteria to two cycles of operation. Over this period, tubes that have degradation in the WEXTEX region with a high potential for leakage based on NDE will be pressure tested (in-situ) in order to acquire data necessary for determining the accuracy of the model. The licensee has established criteria for tube selection specific to Diablo Canyon Units 1 and 2 that are based on industry developed guidance for leakage integrity in-situ pressure testing. If the initial results from testing indicate that tubes are not leaking as expected, the criteria may be modified by the licensee to increase the potential for obtaining leak rate data. Ultimately, the objective of the in-situ pressure testing program is to demonstrate that the W* leakage model yields estimates of leakage that are approximately equal to or less than measured leak rates from a number of service-induced flaws in the WEXTEX region. The requirement to perform in-situ pressure testing is included in the supplement to the steam generator tube surveillance requirements that limits the repair criteria to two cycles of operation.

The W* repair criteria permit all forms of degradation to remain in service provided the indications exist below the lower boundary of the flexible W* distance. The ability to neglect degradation outside the flexible W* distance is a result of the licensee's assessment that demonstrated adequate structural integrity of a tube is maintained without considering the reinforcing effect provided by the length of tubing below this region. The licensee has also concluded that neglecting the contribution to the total leak rate due to flaws located below the flexible W* distance is reasonable. The total leak rate from tubes repaired by W* is primarily governed by those flaws that exist in near the tubesheet secondary face. The contact between the tube and tubesheet in the WEXTEX expansion provides considerable resistance to primary-to-secondary leakage. This resistance increases as flaws are located further into the tubesheet expansion. On this basis, the staff concludes that the determination of a total leak rate based only on those flaws existing within the flexible W* distance is acceptable.

The accuracy of the leak rate model used for W* repairs cannot be described in quantitative terms at this time due to a lack of data from service-induced degradation in actual steam generator tubing. However, the model was developed considering the appropriate factors that are expected to affect leakage from PWSCC in WEXTEX expansions. Empirical constants within the model were based on actual testing under conditions simulating inservice tubes. The total leak rate determined in the operational assessment will be calculated using appropriately conservative statistical intervals. In addition, operational experience has demonstrated that PWSCC in WEXTEX tubesheet expansions has a low likelihood for leakage during normal operation. Therefore, it is expected that the potential for leakage under accident conditions will also remain low. Finally, the Diablo Canyon steam generators have experienced only moderate levels of degradation in WEXTEX expansions to date. The limited duration of this amendment should ensure that the number of steam generator tubes affected by PWSCC remains low at these units over this period of time. On these bases, the staff concludes that the W* repair criteria will provide adequate margins for steam generator tube leakage integrity.

3.3 Nondestructive Examination Techniques

For degraded tubes repairable through the application of the W* criteria, the location of the degradation as well as its orientation (i.e., axial, circumferential, volumetric) and geometry must be assessed. Specifically, the essential objectives for implementing W* are as follows: (1) the determination of the number of flaws and their location with respect to the BWT, (2) confirmation that the flaw(s) is a predominantly axially-oriented crack, (3) the measurement of the overall length of degraded regions in the WEXTEX expansion, (4) the inspection of a sufficient length of expanded tube to ensure adequate structural integrity (i.e., flexible W* length), and (5) an understanding of the uncertainties involved in the NDE inspection process. The long-term condition of W* tubes is maintained, in part, by reanalyzing known tubesheet flaws at each inservice inspection to verify that the assumptions of the repair criteria remain valid. WCAP-14797, Revision 1, documents the qualification of NDE techniques for each of the essential tasks.

The detection of flaws in and near the tubesheet expansion-transition is accomplished through traditional inspection techniques employed by the industry for detecting flaws in this area. The requirements for these inspections are governed by criteria presently included in the plant technical specifications. In general, licensees establish inspection scopes for tubesheet examinations that significantly exceed the minimum inspection requirements included in the technical specifications. No augmented inspection techniques for flaw detection are mandated for the implementation of W* since these repair criteria are employed only after cracks are identified. This practice is consistent with other approved alternate steam generator tube repair criteria for degradation in the tubesheet region.

Flaws located in the tubesheet expansion region must first be characterized using appropriate inspection methods in order to determine whether the W* repair criteria apply. Degradation detected in this area is generally the result of axial or circumferential cracking or volumetric intergranular attack. The W* repair criteria apply only to predominantly axially-oriented cracks located below the BWT. Therefore, all tubes containing circumferential cracks and intergranular attack degradation in the flexible W* distance must be removed from service. Tubes containing flaws of any morphology located below the flexible W* distance may remain in service. The primary inspection technique that would be used to characterize the steam generator tube degradation is an examination using rotating eddy current probes. Previous inspections of steam generator tube defects using such probes have demonstrated the ability to distinguish between these three flaw geometries. Therefore, inspections completed for W* tube repairs should enable the accurate characterization of tubesheet degradation.

Although available tubesheet degradation data suggest that flaws are generally of one of the three orientations listed previously, the possibility exists that cracks could initiate and grow at intermediate angles with respect to the axial or circumferential axes of the tube. The structural criteria established for the W* repair criteria require all flaws to have crack angles of less than 45 degrees from the axial direction. In order to minimize the possibility of returning tubes to service with flaws exceeding this criterion, the licensee quantified the uncertainty associated with measuring the angle of inclination of tubesheet flaws. Using these results, the maximum allowable angle for inservice degradation was reduced by the 95 percent upper confidence limit on the uncertainty of the angle measurement.

One of the primary objectives necessary for successful implementation of W* is the accurate positioning of the tips of any cracks identified in the tubesheet expansion within the flexible W* length. The licensee will record tubesheet expansion crack locations with respect to the BWT. Because measurements are generally made with respect to the tubesheet secondary face, it was necessary to quantify the combined uncertainty in determining the position of the BWT and the crack tip. W* inspections will incorporate technology to ensure a high degree of accuracy in measuring distances relative to the BWT. The accuracy of the distance measurements is critical to determining the accident-induced leak rate from flaws permitted to remain in service and to ensure adequate structural integrity by establishing an appropriate flexible W* distance. The licensee completed a qualification program to estimate the error associated with performing the length measurements of W*. Based on the staff's review of the qualification, the inspection procedures provide an adequate process for ensuring the criteria of W* are maintained.

Upon returning degraded tubes to service using the W* repair criteria, new flaws may initiate and existing cracks may continue to propagate during subsequent operating cycles. Crack growth during operation may lead to a state where the degradation in a tube exceeds the assumptions in the W* analyses. To accommodate such changes, the licensee will employ appropriate allowances for uncertainties in measuring crack lengths and positions and additional margins for crack growth in establishing the flexible W* length. Therefore, inspecting over the flexible W* length should ensure adequate tube integrity exists at the end of the next operating cycle. These margins were developed during the qualification process for the proposed inspection techniques. At each refueling outage, the licensee is required to reinspect all tubes returned to service using W*. Therefore, the condition of degradation in these tubes will be routinely reassessed to minimize the potential of operating with tubes outside the conditions assumed in W*.

Implementation of the W* repair criteria is primarily dependent on accurately acquiring and appropriately analyzing nondestructive examination data. Proper dispositioning of tube indications involves the ability to assess the mode and geometry of identified tube degradation and determine the position of cracks with respect to the BWT. The licensee has completed a qualification program and developed an appropriate methodology to ensure and accurate assessment of W* flaws. In addition, W* tubes will be reinspected at each outage to ensure structural and leakage integrity margins are maintained. The staff concludes that the proposed inspection techniques and the qualification program completed to assess NDE testing errors should ensure that the conditions of W* are accurately assessed and implemented with appropriate margins. Therefore, the staff concludes that the inspection techniques proposed for the W* repair criteria are acceptable.

3.4 W* Reporting Requirements

Following each inservice inspection, the licensee will report to the NRC the results of its examination of tubes repaired using the W* repair criteria and the leak rates calculated from the condition monitoring and operational assessments. If circumstances arise that were not considered in the development of the proposed repair criteria, the NRC would be able to assess whether the licensee can continue to implement W*. The requirement to submit this information will enable the NRC to monitor the effectiveness of the alternate repair criteria to

maintain adequate margins for steam generator tube structural and leakage integrity. On this basis, the staff concludes that the proposed reporting requirements are acceptable.

3.5 Risk Considerations

The W* alternate tube repair criteria are not expected to increase the level of risk and are designed to identify and repair tubes with cracks that are presently below the BWT but which might grow above the tubesheet into the freespan region before the next inspection. Measurements of crack growth rates during the previous cycle will be used to determine the allowance for growth for the next operating cycle. Because this license modification is limited to two fuel cycles, the efficacy of the licensee's program to avoid the propagation of cracks into the freespan can be assessed before a decision is made to make the modification permanent. Avoiding crack propagation into the freespan obviates concerns about the potential for gross failure of the primary-to-secondary boundary due to steam cutting of the preexisting cracks and adjacent tubes during core damage accidents. In addition, the leakage calculations for cracks in portions of the tubes below the tubesheet secondary face are conservatively estimated at the 95% quantile value for design basis accident analyses. The applicable limit at Diablo Canyon is 12.8 gpm for the combination of leakage from the W* and ODSCC at the drilled-hole TSPs. The median leakage rate estimate is expected to be approximately an order of magnitude below the 95% estimate. On that basis, actual leakage during severe accidents is not expected to significantly compromise the performance of the containment function, so the strategy of defense-in-depth against the accidental release of radioactive material is maintained.

3.6 Technical Specification Changes

In order to incorporate the alternate repair criteria for indications in the WEXTEX (W*) region of the steam generators, the licensee has proposed the following changes to the TS.

a. Footnotes to TS 3.4.5

The first footnote limits the duration of the applicability of the TS amendment to the next two cycles (Cycles 10 and 11) of operation for each unit.

The second footnote states that the in-situ testing described in the October 22, 1998 (DCL-98-148) letter is followed.

b. Proposed New TS 4.4.5.2.b.5)

The TS provides a new requirement for examining previously identified W* tubes using a rotating pancake coil probe.

c. Proposed New TS 4.4.5.4.a.6)

The TS changes the current definition of Plugging Limit to account for the W* criteria.

d. Proposed New TS 4.4.5.4.a.11)

The TS provides a new definition for the W* Plugging Limit, associated additional supporting definitions, and acceptable degradation allowed within the W* region.

e. Proposed Change to TS 4.4.5.5.d.1)

A proposed change that clarifies a criterion for notifying the NRC based on estimated leakage.

f. Proposed New TS 4.4.5.5.e and f

The TS adds a new reporting requirement for W* tubes and also one for which has an alternate repair criteria is applied.

g. Proposed Changes to Bases 3/4.4.5

The Bases are revised to add additional information that explains the basis for the W* criteria and the limits of applicability.

h. Administrative Changes

Administrative changes are made to TS 4.4.5.2.b.1), 4.4.5.2.b.2), 4.4.5.4.a.8), 4.4.5.4.a.9), and Bases TS 3/4.4.6.2. These changes clean up the existing TS by removing extraneous words or punctuation. The changes makes the Bases more consistent with GL 95-05.

3.7 Conclusions

The licensee has proposed to amend the Diablo Canyon, Units 1 and 2, technical specifications to permit the application of alternate steam generator tube repair criteria (W*). Based on the staff's evaluation of the repair criteria as documented previously, it will maintain adequate margins for tube structural and leakage integrity in accordance with the regulatory requirements included in Section 2.1 of this safety evaluation. On this basis, the staff concludes that the proposed changes to the Diablo Canyon technical specifications are acceptable. Due to a lack of data, the licensee was unable to demonstrate the conservatism in the W* leak rate model. Therefore, the duration of the amendment was limited to two cycles of operation for each unit. During this interval, the licensee will perform in-situ pressure testing of tubes with degradation in WEXTEX expansions that exceed criteria established in this amendment in order to validate the leak rate model. The licensee may consider proposing the W* repair criteria on a permanent basis once the capabilities of the leak rate model has been quantified.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the California State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (62 FR 61843). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). These amendments also involve changes in recordkeeping, reporting or administrative procedures or requirements. Accordingly, with respect to these items, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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