October 28, 2005

Mr. David H. Oatley, Acting Chief Nuclear Officer
Pacific Gas and Electric Company
Diablo Canyon Power Plant
P.O. Box 56
Avila Beach, CA 93424

SUBJECT: DIABLO CANYON POWER PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: APPROVAL OF PERMANENT USE OF THE W* ALTERNATE REPAIR CRITERIA FOR STEAM GENERATOR TUBES (TAC NOS. MC6409 AND MC6410)

Dear Mr. Oatley:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 182 to Facility Operating License No. DPR-80 and Amendment No. 184 to Facility Operating License No. DPR-82 for the Diablo Canyon Power Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated March 11, 2005, and as supplemented by letter dated August 25, 2005.

The amendments allow use of the steam generator tube W* (W-star) alternate repair criteria for indications in the Westinghouse explosive tube expansion region on a permanent basis.

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,

**/RA/**
Girija S. Shukla, Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-275
and 50-323

Enclosures:
1. Amendment No. 182 to DPR-80
2. Amendment No. 184 to DPR-82
3. Safety Evaluation

cc w/encls: See next page

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| DATE | 10/26/05 | 10/26/05 | 9/22/05 | 10/27/05 | 10/28/05 |

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PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-275

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 182
License No. DPR-80

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

   A. The application for amendment by Pacific Gas and Electric Company (licensee) dated March 11, 2005, and as supplemented by letter dated August 25, 2005, comply 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. 182, 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 shall be implemented during steam generator inspections conducted in Unit 1 Refueling Outage 13 (1R13).

FOR THE NUCLEAR REGULATORY COMMISSION

**/RA/**
Daniel S. Collins, Acting Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 28, 2005

PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-323

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.184
License No. DPR-82

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

   A. The application for amendment by Pacific Gas and Electric Company (licensee) dated March 11, 2005, and as supplemented by letter dated August 25, 2005, comply 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. 184, 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 shall be implemented during steam generator inspections conducted in Unit 2 Refueling Outage 13 (2R13).

FOR THE NUCLEAR REGULATORY COMMISSION

**/RA/**
Daniel S. Collins, Acting Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 28, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 182

TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 184 TO FACILITY OPERATING LICENSE NO. DPR-82

DOCKET NOS. 50-275 AND 50-323

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

| REMOVE | INSERT |
|---|---|
| 5.0-10 | 5.0-10 |
| 5.0-13 | 5.0-13 |
| 5.0-13a | 5.0-13a |
| 5.0-15 | 5.0-15 |
| 5.0-16 | 5.0-16 |
| 5.0-17 | 5.0-17 |
| 5.0-29 | 5.0-29 |
| 5.0-30 | 5.0-30 |
| 5.0-30a | 5.0-30a |
| ----- | 5.0-30b |

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 182 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 184 TO FACILITY OPERATING LICENSE NO. DPR-82

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON POWER PLANT, UNITS 1 AND 2

DOCKET NOS. 50-275 AND 50-323

## 1.0 INTRODUCTION

By application dated March 11, 2005, and as supplemented by letter dated August 25, 2005 (Agencywide Documents Access and Management System (ADAMS) accession numbers ML050750134 and ML052440396, respectively), Pacific Gas and Electric Company (PG&E or licensee) requested changes to the Technical Specifications (TS; Appendix A to Facility Operating License Nos. DPR-80 and DPR-82) for the Diablo Canyon Power Plant, Units 1 and 2 (DCPP). The supplemental letter dated August 25, 2005, 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 as published in the *Federal Register* on April 26, 2005 (70 FR 21462).

The requested change would revise the DCPP TS to allow use of the steam generator (SG) tube W* (W-star) alternate repair criteria (ARC) on a permanent basis. Specifically, the proposed amendment would revise DCPP TS 5.5.9, "Steam Generator Tube Surveillance Program," and TS 5.5.10, "Steam Generator Inspection Report." This W* ARC allows axial primary water stress corrosion cracking (PWSCC) in the Westinghouse explosive tube expansion (WEXTEX) region to remain in service provided the indication remains below the bottom of the WEXTEX transition (BWT) during the next operating cycle. The length of the tube required to be inspected within the hot-leg tubesheet is referred to as the W* distance or flexible W* distance.

The proposed changes would:

1. Remove the existing note "Applicable for Units 1 and 2, Cycles 10, 11, 12 and 13 only" for TSs 5.5.9.b.2.e, 5.5.9.d.1.f.2, 5.5.9.d.1.k, 5.6.10.d.1, 5.6.10.e, and 5.6.10.f,

2. Replace a period with a comma at the end of TS requirement 5.5.9.b.2.d,

3. Terminate the current W* in-situ testing program by removing the existing note "In-Situ Testing will be performed in accordance with PG&E Letters DCL 98-148 dated October 22, 1998, and DCL 01-0152 dated May 4, 2001, for Cycles 10 and 11, and Letter DCL 01-095 dated September 13, 2001, for Cycles 12 and 13,"

4. Revise TS 5.5.9.d.1.k.ii to update the reference from “WCAP-14797, Revision 1” to “WCAP-14797-P, Revision 2,”

5. Revise TS 5.6.10.d.1 to correct a typographical error and to indicate the Nuclear Regulatory Commission (NRC) notification requirement for voltage-based repair criteria is based on the W* leakage *increased* by estimated leakage by all other ARC**,**

6. Delete the TS 5.6.10.d.2 requirement “If circumferential crack-like indications are detected at the tube support plate intersections,”

7. Delete the TS 5.6.10.d.4 requirement “If indications are identified at tube support plate elevations that are attributable to primary water stress corrosion cracking,”

8. Renumber TS 5.6.10.d.3 to TS 5.6.10.d.2 and include “ODSCC” to clarify the reporting requirement only pertains to ODSCC indications that are applicable to a voltage-based repair criterion; renumber TS 5.6.10.d.5 to TS 5.6.10.d.3,

9. For TS 5.6.10.e, replace Requirements 1 through 4 with a new expanded Requirement 1 containing the W* inspection information to be reported within 90 days following return to service of the SGs,

10. For TS 5.6.10.e, add a new Requirement 2 to state “Assessment of whether the results were consistent with expectations and, if not consistent, a description of the proposed corrective action,”

11. Revise TS 5.5.9.d.1.k (v).6 to state that any type or combination of tube degradation below the flexible W* length is acceptable,

12. Revise TS 5.5.9.b.2.e to add the words “flexible” and “or equivalent” to clarify the inspection distance to be accomplished and to acknowledge that probe technology other than rotating pancake coil probes may be used for inspection, provided these probes are capable of detecting the degradation that may be present, as discussed in Generic Letter 2004-01,

13. Revise TS 5.5.9.d.1.k (ii) to state the W* length is the distance *in* the tubesheet below the BWT, and clarify 5.5.9.d.1.k (iv) by deleting the words “equal or greater than 40%.”

## 2.0 REGULATORY EVALUATION

Because of the importance of SG tube integrity, the NRC requires the performance of periodic inservice inspection (ISI) of SG tubes. These inspections detect, in part, degradation in the tubes as a result of the SG operating environment. ISIs may also provide a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken. Tubes with degradation that exceeds the tube repair limits specified in a plant's TSs are removed from service by plugging or are repaired by sleeving (if approved by the NRC for use at the plant). The plant TSs provide the acceptance criteria related to SG tube inspections.

The requirements for the inspection of SG tubes are intended to ensure that this portion of the reactor coolant system maintains its structural and leakage integrity. Structural integrity refers to maintaining adequate margins against gross failure, rupture, and collapse of the SG tubes. Leakage integrity refers to limiting primary-to-secondary leakage during normal operation, plant transients, and postulated accidents to ensure that the radiological dose consequences are within acceptable limits.

In reviewing requests of this nature, the NRC staff verifies that the structural and leakage integrity of the tubes will continue to be maintained consistent with the plant design and licensing basis. This includes verifying that the applicable General Design Criteria (GDC) (e.g., GDC 14 and 32) of Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, are satisfied and that the structural margins inherent in Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR [pressurized-water reactor] Steam Generator Tubes," dated August 1976, are maintained. This also includes verifying that a conservative methodology exists for determining the amount of primary-to-secondary leakage during design-basis accidents. The amount of leakage is limited to ensure that offsite and control room dose criteria are met. The radiological dose criteria are specified, in part, in 10 CFR Part 100 and in GDC 19 of Appendix A to 10 CFR Part 50.

The NRC previously approved for two cycles, and renewed for two additional cycles, similar but not identical W* ARC amendment requests for DCPP in Amendment Nos. 129/127, dated February 19, 1999 (NUDOCS accession number 9903030010), and Amendment Nos. 151/151, dated April 29, 2002 (ADAMS accession number ML021200166). In addition, the NRC approved a similar request for Sequoyah Nuclear Plant, Unit 2 (ADAMS accession number ML021340595) in License Amendment No. 266 for one cycle of operation. More recently, the NRC approved a similar request for Beaver Valley, Unit 1 (ADAMS accession number ML042730591) in License Amendment No. 262 for one cycle of operation and a permanent W* amendment for Sequoyah Nuclear Plant, Unit 2 (ADAMS accession number ML051160009) in License Amendment 291.

## 3.0 TECHNICAL EVALUATION

As discussed above, the proposed amendment would revise the DCPP TS to allow use of the steam generator tube W* ARC on a permanent basis. In addition, the proposed amendment would eliminate the need for additional in-situ pressure testing of indications within the tubesheet. This amendment will also eliminate some of the NRC notifications that are no longer necessary. Finally, the amendment contains several administrative changes such as correcting a typographical error or making minor editorial changes to clarify the meaning of the W* length or to revise a reference.

### 3.1 Background

DCPP Units 1 and 2 are 4-loop, Westinghouse-designed plants containing Model 51 SGs. Each SG contains about 3400 tubes. The SG tubes are mill-annealed Alloy 600 with an outside diameter of 0.875 inch and a wall thickness of 0.050 inch. Each U-tube is roll-expanded for approximately 2.75 inches into the bottom of the tubesheet, then secured into the remaining portion of the tubesheet by an explosive expansion process referred to as the WEXTEX process.

The tubesheet is approximately 21 inches thick and each tube is expanded for essentially the full thickness of the tubesheet. Each tube is also welded to the primary side of the tubesheet near the tube end. This weld provides a leak tight boundary and also provides resistance to tube pullout. The WEXTEX process forms an interference fit between the tube and tubesheet. The transition from the expanded portion of the tube to the unexpanded portion of the tube is referred to as the WEXTEX transition or the expansion transition. Each SG contains seven tube support plates to provide lateral support to the tubes. The tube supports are carbon steel plates with drilled holes through which the tubes are inserted.

The original plant TSs did not take into account the reinforcing effect of the tubesheet on the external surface of the expanded tube. The presence of the tubesheet constrains the tube and complements tube integrity in that region by essentially precluding tube deformation beyond the expanded outside diameter of the tube. The resistance to both tube rupture and tube collapse is significantly enhanced by the tubesheet reinforcement. In addition, the proximity of the tubesheet to the expanded tube significantly reduces the leakage from any through-wall defect.

Based on these considerations, power reactor licensees have proposed, and the NRC has approved, ARC for defects located in the SG tube contained in the lower portion of the tubesheet, when these defects are a specific distance below the expansion transition or the top-of-the-tubesheet (TTS), whichever is lower.

The W* methodology defines a distance, referred to as the W* distance, such that any type or combination of tube degradation below this distance is considered acceptable (i.e., even if inspections below this region identified degradation, the regulatory requirements pertaining to tube structural and leakage integrity would be met provided there were no flaws within the W* distance). The W* distance is determined by calculating the amount of undegraded tubing, termed the W* length, needed to address tube pullout and leakage concerns. This W* length is measured from the bottom of the WEXTEX transition region. One of the key considerations in determining the W* length is the amount of undegraded tubing necessary to prevent an axial pullout of the tube from the tubesheet. Tube pullout could result from the internal pressure in the tube. In addition to the W* length, nondestructive examination (NDE) uncertainties must be accounted for to determine the W* distance. These uncertainties include, but are not limited to, the uncertainties in determining the location of the bottom of the WEXTEX expansion and the total inspection distance below this point (i.e., W* length). These uncertainties are addressed in the W* methodology and were discussed in a previous NRC staff's safety evaluation for DCPP Amendment Nos. 129/127, dated February 19, 1999 (NUDOCS accession number 9903030010) approving the W* ARC for two cycles.

The generic W* analysis presented in WCAP-14797, Revision 2, uses bounding or non-plant-specific values for secondary system pressure and primary temperature to determine the required W* length for two regions of the tube bundle. This analysis considered the forces acting to pull the tube out of the tubesheet (i.e., from the internal pressure in the tube) and the forces acting to keep the tube in place. These latter forces are a result of friction and the forces arising from (1) the residual preload from the WEXTEX expansion process, (2) the differential thermal expansion between the tube and the tubesheet, and (3) internal pressure in the tube within the tubesheet. In addition, the effects of tubesheet bow due to pressure and thermal differentials across the tubesheet were considered since this bow causes dilation of the tubesheet holes from the secondary face to approximately the midpoint of the tubesheet and reduces the ability of the tube to resist tube pullout. The amount of tubesheet bow varies across the tube bundle with tubes in the periphery (referred to as Zone A tubes) experiencing less bow than tubes in the interior of the SG tube bundle (referred to as Zone B tubes). In fact, the analysis provided indicates that the W* length for Zone A tubes is 5.2 inches and the W* length for Zone B tubes is 7.0 inches. In addition to the W* length, the analysis in WCAP-14797 also considered the uncertainties associated with NDE.

In addition to the W* length discussed above, the licensee has proposed tube inspections to a minimum depth of 8.0 inches below the top-of-the-tubesheet (TTS) to be consistent with their leakage integrity analysis.

3.2 Diablo Canyon's Proposal

The licensee's basis for a permanent amendment related to a W* ARC is documented in its license amendment request and in WCAP-14797, Revision 2. Operating conditions assumed in the generic WCAP analysis bound the operating conditions at DCPP such that the W* distance calculated using plant-specific conditions would be less than the W* distance identified in WCAP-14797. For example, the generic analysis assumes a hot-leg temperature of 590 $^{o}$F whereas the limiting hot-leg operating temperature at DCPP is significantly higher. Therefore, the generic analysis provides less thermal tightening of the WEXTEX joint than would actually be present in the DCPP SGs. The secondary system pressure assumed for the generic analysis also provides for less pressure tightening of the WEXTEX joint compared to the plant conditions.

The proposed amendment uses a leakage methodology that combines a constrained crack leakage model and a severed tube leakage model that are based on tube-to-tubesheet contact pressures. This leakage methodology is more conservative than the DENTFLO Code leakage model presented in WCAP-14797 and used for the existing DCPP two-cycle W* amendment.

The following sections summarize the NRC staff's evaluation of the proposed W* proposal in terms of maintaining SG structural and leakage integrity.

3.3 Tube Structural Integrity

The NRC staff has previously reviewed and approved the structural integrity component of the W* ARC related to DCPP. The confinement of the surrounding tubesheet for all flaws left in service using this proposed ARC will prevent tube structural failure by tube burst. Use of the W* ARC will also ensure that tube pullout from the tubesheet under the limiting conditions is precluded. Therefore, the NRC staff concluded that tubes returned to service using the W* ARC will maintain adequate structural integrity.

3.4 Tube Leakage Integrity

In assessing leakage integrity of a SG under postulated accident conditions, the leakage from all sources (i.e., all types of flaws at all locations and all non-leak tight repairs) must be assessed. The combined leakage from all sources is limited to below a plant-specific limit determined based on radiological dose consequences. The licensee's plant-specific limit is currently 10.5 gallons per minute (gpm) in any one SG. For flaws located within the tubesheet, the licensee has developed a methodology to determine the total amount of accident-induced primary-to-secondary leakage. This methodology determines leakage from indications within the W* distance, (i.e., approximately 7.1 inches below the bottom of the WEXTEX transition) or flexible W* distance, if applicable. In addition, since the W* methodology does not require inspections below the W* distance (or flexible W* distance), there is a potential that flaws that could leak will exist below the W* distance. As a result, the licensee also determines leakage from flaws in this region of the tubesheet. The licensee's leakage methodology and the NRC staff's review of this methodology are discussed below.

The W* ARC previously approved for DCPP for Cycles 10, 11, 12, and 13 calculates accident induced leakage using a model referred to as the DENTFLO Model. The DENTFLO model determines the leak rate by integrating two empirical relationships: leak rate through a constrained crack and leak rate through a tube-to-tubesheet crevice. Previous NRC approval for the DCPP W* ARC was limited to two cycles due to NRC questions about the DENTFLO leakage model validity. Therefore, the licensee proposed renewing the W* ARC on a permanent basis using a more conservative leakage methodology. The new leakage methodology uses both a constrained crack model (for leakage from indications located between the TTS and TTS minus 12 inches) and a severed tube model (for leakage from indications below TTS minus 12 inches). With the new methodology, total leakage from indications within the tubesheet will be determined by summing: (1) leakage from indications within the W* (or flexible W*) distance, (2) leakage from <u>detected</u> indications between the W* (or flexible W*) distance and TTS minus 12 inches, (3) leakage from <u>undetected</u> indications between the W* (or flexible W*) distance and TTS minus 12 inches, and (4) leakage from degradation below TTS minus 12 inches.

This methodology requires a determination of the total number of indications within the tube increments identified above and the leak rate from these indications. The total number of indications within the W* distance is determined by inspection. In addition, since actual inspection lengths are greater than the W * inspection distance, some indications between the W* length and TTS minus 12 inches may be detected by inspection. Therefore, the number of indications within items (1) and (2) above are determined by inspection. The licensee estimated the number of indications from <u>undetected</u> indications between the W* (or flexible W*) distance and TTS minus 12 inches (item 3 above) using two methods. In the first method, historical inspection data between TTS minus 4 inches and TTS minus 8 inches was used to project the number of indications between TTS minus 8 inches and TTS minus 12 inches. This approach was judged to be conservative since inspection data showed a decreasing number of indications with distance below the TTS. A second method of estimating the number of indications involved fitting a regression line to the inspection data (excluding the expansion transition region) and projecting the number of indications between TTS minus 8 inches and TTS minus 12 inches using a 90-percent probability prediction bound. The licensee chose this latter method since it projects a greater number of indications between the W* distance and TTS minus 12 inches. To estimate the number of indications below TTS minus 12 inches, the licensee conservatively assumes all tubes remaining in service contain a 360-degree circumferential, 100 percent through-wall flaw (i.e., a tube sever) located 12 inches below TTS.

After determining the number of indications located within the tubesheet, the total leakage from those indications is obtained by multiplying the total number of indications by the appropriate leak rates. The licensee’s leakage calculations rely on relationships developed from the data provided in WCAP-14797. For those indications located from TTS to minus 12 inches below TTS, the licensee determines leakage with the constrained crack leak model. Constrained crack leak tests were performed on representative WEXTEX test specimens that were designed to provide a leak path with minimal resistance from the tube-to-collar crevice.
Two different specimen collar sizes were used to obtain different initial diametral gaps between the tube and collar. Once these initial test conditions were established, temperature and

pressure were changed to develop different levels of contact pressure. Constrained crack leakage was then determined as a function of contact pressure between the tube and collar.

The licensee's constrained crack leakage methodology (employed for indications from TTS to 12 inches below TTS) employs a 95$^{th}$ percentile prediction bound on the leak rate as a function of contact pressure for the full set of constrained crack leak data. Contact pressures in the steam generator are determined from finite element analysis, based on the tube radial position and indication depth within the tubesheet. Thus, for all detected indications, leak rates are assigned using a 95$^{th}$ percentile prediction bound given the contact pressure at the location of the flaw. For the projected set of undetected indications between the W* distance and 12 inches below TTS, the licensee assigns a constrained crack leak rate (3.3 x $10^{-3}$ gpm) corresponding to the SG zone with the minimum contact pressure at 8 inches below TTS.

During the NRC staff's review of the licensee's constrained crack leakage methodology, the staff noted a higher variance in the population of constrained crack leak rate data with contact pressure above 1200 psi. The NRC staff concluded that the constrained crack leak data consisted of two data sets that could be separated by a contact pressure of approximately 1200 psi. Thus, the NRC staff considers the appropriate statistical treatment of the data would involve fitting regression lines to each population of data and extrapolating the regression line from the lower contact pressure data set to a higher contact pressure. Although, the NRC staff does not agree with the licensee's methodology that assumes one population of data, the resulting numerical leak rate values used by the licensee were more conservative (i.e., higher leak rates) than those obtained by treating the data as two populations. The licensee's methodology is particularly conservative for the lower contact pressures data, which represents those indications closer to the TTS that have a higher likelihood of leaking. Therefore, the NRC staff finds the numerical values of the constrained crack leak rates assigned by the licensee acceptable.

For indications below TTS minus 12 inches, the licensee determines leakage using a severed tube model. The licensee conservatively assumes all in-service tubes contain a 360-degree tube sever located 12 inches below the TTS and assigns an 9 x $10^{-5}$ gpm leak rate per tube. Use of a severed tube model for indications below TTS minus 12 inches was previously reviewed and approved by the NRC staff for use at Beaver Valley Unit 1 (ADAMS accession number ML042730587). Given past plant-specific and industry operating experience, the NRC staff considers the assumption that all tubes contain circumferential, through-wall flaws at 12 inches below TTS, along with application of an upper 90 percent leak rate to these flaws, to be conservative given the current condition of the SGs.

The licensee is making an additional change in the W* ARC related to disposition of indications near the bottom of the WEXTEX transition (BWT). Any indications found above the bottom of the BWT or projected to be above the BWT in the next cycle, after including NDE uncertainty and crack growth allowance, are repaired. The NRC staff finds this acceptable since this will provide reasonable assurance all cracks are fully constrained within the tubesheet (i.e., the tube is fully expanded against the tubesheet).

Based on a review of this information, the NRC staff considers the leak rates from the licensee's methodology acceptable. A number of conservative assumptions have been included in this leakage methodology:

1. All indications within the tubesheet are assumed to be through-wall. Historical inspection data indicates only a fraction of these indications are through-wall. Thus, this is a conservative assumption.

2. The constrained crack leak rates (applied to indications from the TTS to minus 12 inches) are based on tests simulating the resistance to crack leakage provided by the tubesheet constraining the crack opening. In reality, leakage from tubes with through-wall flaws located within a tubesheet in the SG is also restricted by the crevice (or more precisely the interference fit) between the tube and tubesheet. The licensee's leakage methodology takes no credit for the leakage restriction resulting from the tube-to-tubesheet crevice.

3. Stress corrosion cracks in SG tubes tend to be tighter than the fatigue cracks generated in the constrained crack test samples, thereby providing greater resistance to leakage.

4. The licensee conservatively assumes all tubes in service contain a 360-degree tube sever located 12 inches below the TTS and therefore each in-service tube contributes to the leakage total.

The NRC staff also compared the relative conservatism between the licensee's existing W* leak methodology (based on the DENTFLO model) and the new W* leak rate methodology (constrained crack and severed tube models), using the projected Operation Assessment leak rates for Units 1 and 2, End of Cycle (EOC) 13. For the Unit 1 SGs, the projected leak rates using the new leakage methodology were approximately 10 times to 60 times greater than those obtained with the DENTFLO leakage model. Projected EOC accident-induced leak rates in the Unit 2 SGs were approximately four times to seven times greater with the new W* leak rate methodology.

The licensee's application of the W* ARC and accompanying leakage methodology will be used to determine the amount of leakage from flaws within the tubesheet, either within or below the W* distance (or flexible W* distance). This leakage will be combined with the leakage from all other sources to ensure that it is less than the plant-specific allowable limits. In addition, the licensee will be required to assess whether the results of the inspection were consistent with expectations with respect to the number of flaws and their severity. In the event that the results are not consistent, the licensee will be required to describe proposed corrective actions. On this basis, the NRC staff has concluded that the licensee has an acceptable methodology for assessing leakage to address degradation within the tubesheet region, thereby ensuring leakage integrity can be maintained.

3.5 Reporting Requirements

Within 90 days of returning the SGs to service (Mode 4), the licensee will report the following with respect to implementation of the W* inspection methodology:

- Identification of W* tube indications and other indications that do not meet W* requirements and were plugged or repaired.

- The number of indications, the location of the indications (relative to the BWT and TTS), the orientation, the radial position of the tube within the tubesheet, the W* zone, and the severity of each indication.

- Degradation initiation with respect to the tube inside diameter or outside diameter.

- The W* inspection distance, the length of axial indications, the angle of inclination of skewed axial cracks, and verification that the upper crack tip of axial indications returned to service in the prior cycle remain below the BWT by at least the 95 percent confidence NDE uncertainty on locating the crack tip relative to the BWT.

- Updated 95 percent growth rates for use in the 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 Letters DCL-05-018 and DCL-05-090 dated March 11, 2005 and August 25, 2005, respectively.

- In addition, an assessment will be performed to determine if the results were consistent with expectations and, if not consistent, a description of proposed corrective action will be provided.

3.6 Other Changes

In addition to the changes mentioned above, the licensee also proposed several additional changes. One change would terminate the current W* in-situ testing program and only in-situ pressure test those indications that meet the requirements of the Electric Power Research Institute In-Situ Guidelines. In-situ pressure testing of W* indications was initiated in response to questions concerning the validation of the WCAP-14797 DENTFLO leakage model. Since the licensee's new leakage model is more conservative than the DENTFLO model, in-situ pressure testing is no longer needed and the NRC staff finds this change acceptable. The NRC staff notes that although no leakage has been reported for a total of 21 tests performed at Diablo Canyon Unit 2, Sequoyah Unit 2 and Beaver Valley Unit 1, in-situ pressure testing is not able to reproduce certain accident induced effects, such as tubesheet hole dilation resulting from tubesheet bow.

This amendment will also eliminate some of the NRC notifications related to detailed PWSCC ARC reporting criteria. This is acceptable to the NRC staff since they are no longer necessary. The amendment also contains several other changes that are either intended to improve the TS clarity or are administrative. Examples include adding additional information in the reporting requirements, correcting a typographical error, minor editorial changes intended to clarify the meaning of the W* length, or changes to revise a reference. The NRC staff finds these changes acceptable.

3.7 Summary

The NRC staff's approval of the licensee's proposal is based on the licensee demonstrating that applicable structural integrity and leakage integrity requirements will be met. This approval is based, in part, on inspections and conservative assumptions involving the licensee's implementation of the W* ARC including:

8. The licensee is performing inspections to a minimum depth of 8 inches below the TTS and repairing all indications that could grow above the BWT during the next operating cycle, including allowances for NDE uncertainty and crack growth. These inspections are required to detect the forms of degradation occurring in this region.

9. The generic W* distances were determined using bounding parameters (i.e., secondary-side pressure and primary-side temperature) resulting in more conservative W* distances than would be obtained using plant-specific operating parameters. The generic W* distances were also determined from lower bound tube pull forces for WEXTEX expansions (based on a smooth tubesheet hole) in order to maximize the W* distance and bound the variability in WEXTEX expansions.

10. The most limiting region of the tube bundle is Zone B, which is in the interior of the tube bundle. If tubes in this region began to pull out of the tubesheet, they would be constrained by contact with neighboring tubes such that they would not pull out of the tubesheet. As a result, the likelihood that a tube would pull out of the tubesheet is small. This effect was not considered in the development of the W* distance and adds conservatism to the evaluation.

11. The licensee's tubes are most likely experiencing denting at the tube support plates which would further restrain tube pullout and would likely prevent the axial pressure load necessary to cause tube pullout. This effect was not considered in the development of the W* distance and adds conservatism to the evaluation.

12. The licensee projects all postulated indications between 8 inches and 12 inches below the TTS are circumferential, 100 percent through-wall over 360 degrees, and occur in one SG, which is a conservative assumption.

13. The licensee assumes all tubes remaining in service contain a 360-degree circumferential, 100 percent through-wall flaw located 12 inches below the TTS. This assumption is conservative given industry inspection results within the tubesheet region.

14. Flaws postulated below the W* distance are assumed to be leaking although industry operating experience has demonstrated negligible leakage under normal operating conditions, even when cracks are located in a tube-to-tubesheet expansion transition zone.

15. The constrained crack leak rates (applied to indications from the TTS to minus 12 inches) are based on tests simulating the resistance to crack leakage provided by the tubesheet constraining the crack opening. In reality, leakage from tubes with through-wall flaws located within a tubesheet in the SG is also restricted by the crevice (or more precisely the interference fit) between the tube and tubesheet. The licensee's leakage methodology takes no credit for the leakage restriction resulting from the tube-to-tubesheet crevice.

3.8 Conclusion

The NRC staff concludes that the licensee’s proposed methodology for assessing structural and leakage integrity for flaws in the tubesheet region is acceptable. Therefore, the NRC staff concludes that the licensee’s proposal to limit the tube inspection scope in the hot-leg tubesheet is an acceptable approach. In addition, the NRC staff concludes that the other changes being proposed to the technical specifications (e.g., remove some NRC notifications, eliminate the need for additional in-situ pressure testing, and administrative changes) are acceptable.

## 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

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 (70 FR 21462; published April 26, 2005). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). 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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