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PG&E Letter DCL-07-003

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

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

References:
1. PG&E Letter DCL-06-061, "License Amendment Request (LAR) 06-04, Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity (TSTF-449)," dated May 30, 2006
2. PG&E Letter DCL-06-130, "Response to NRC Request for Additional Information Regarding License Amendment Request 06-04, 'Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity (TSTF-449),'" dated November 22, 2006

Dear Commissioners and Staff:

Pacific Gas and Electric Company (PG&E) submitted License Amendment Request (LAR) 06-04, "Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity (TSTF-449)," in Reference 1, and submitted a response to an NRC request for additional information in Reference 2. LAR 06-04 proposes to revise Technical Specification (TS) requirements related to steam generator (SG) tube integrity.

On December 13, 2006, the NRC staff requested additional information required to complete the review of LAR 06-04. PG&E's responses to the staff's questions are provided in Enclosure 1. Enclosure 2 provides revised marked-up TS Bases pages. The TS Bases pages contained in Enclosure 2 supersede the pages provided in Reference 2.

The TS Bases changes clarify the accident analyses assumptions and limits for primary to secondary leakage following design basis events.

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This information does not affect the results of the technical evaluation or the no significant hazards consideration determination previously transmitted in Reference 1.

PG&E makes no regulatory commitments (as defined by NEI 99-04) in this letter. This letter includes no revisions to existing regulatory commitments.

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

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

Executed on January 11, 2007.

Sincerely,

PJR for JR Becker

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

kjse/4328
Enclosures

cc: Edgar Bailey, DHS
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ENCLOSURE 1

Response to NRC Request for Additional Information Regarding License Amendment Request 06-04, "Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity (TSTF-449)"

1. *On page 73 of your Bases (Reactor Coolant System Operational Leakage), you proposed to change the description of your steam generator tube rupture (SGTR) accident. In particular, you proposed to indicate that the power operated relief valve (PORV) fails open for 30 minutes at which time the operator closes the block valve to the PORV. You previously had (in your Bases) that the PORV was assumed to be open for 30 minutes at which time the reactor coolant system pressure was below the lift setting of the PORV. Please confirm that the proposed change is consistent with your currently approved design and licensing basis. If it isn't, please justify the proposed change. The staff notes that similar text is included in the Steam Generator Tube Integrity Bases Section (under Applicable Safety Analyses).*

   PG&E Response:

   The proposed Technical Specifications (TS) Bases change is consistent with the currently approved design and licensing basis assumption documented in the current Final Safety Analysis Report Update (FSARU) Chapter 15.4.3, "Steam Generator Tube Rupture (SGTR)," accident analysis. The analysis assumes the limiting single failure is the failure of the power operated relief valve (PORV) on the ruptured steam generator (SG) in the open position when the ruptured SG is isolated, and that the PORV is isolated by locally closing the associated block valve 30 minutes after the PORV was assumed to fail open. The current FSARU Chapter 15.4.3 SGTR analysis was reviewed by the NRC as part of the NRC approval of revised SGTR radiological consequences analyses in Amendment 156 to Facility Operating License No. DPR-80, and Amendment No. 156 to Facility Operating License No. DPR-82, for Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, respectively, on February 20, 2003.

2. *In several places in your proposed Bases for the Reactor Coolant System Operational Leakage and Steam Generator Tube Integrity (i.e., Insert 1, Applicable Safety Analyses section of the Steam Generator Tube Integrity Bases), you indicate that the primary to secondary leakage during a steam line break (SLB) is assumed to be 10.5 gallons per minute (gpm) or is assumed to increase to 10.5 gpm as a result of accident induced conditions. Although this is consistent with TSTF-449, the text associated with a similar sentence for the accident analyses for other accidents (other than a SGTR and SLB) is not. As a result, please discuss your plans to indicate that: "The safety analyses for events resulting in steam discharge to the atmosphere,*

other than SGTR or SLB, assume that primary to secondary LEAKAGE from all steam generators is 0.75 gpm (hot conditions) or increases to 0.75 gpm as a result of accident induced conditions." The underlined text indicates the potential that the primary to secondary leakage observed during operation may increase under accident conditions.

PG&E Response:

The safety analyses for SLB, locked rotor, and rod ejection events assume a pre-accident leak rate of 1 gpm total from all SGs to set the secondary coolant activity. After the SLB event, the postaccident leakage is assumed to increase to 10.5 gpm in the faulted SG. After the locked rotor and rod ejection events, the postaccident leakage is assumed to be 0.75 gpm total from all SGs, which is less than the assumed pre-accident leakage of 1 gpm total from all SGs. Therefore, for the locked rotor and rod ejection events, the assumed leakage in the analyses does not increase after the event, and the words, "or increases to 0.75 gpm as a result of accident induced conditions," is not considered to be applicable. However, the actual leak rate during operation (i.e., the pre-accident leak rate) will not exceed approximately 0.4 gpm (i.e., 4 times 150 gallons per day [gpd]) based on the TS 3.4.13.d reactor coolant system operational leakage limit of 150 gpd primary to secondary leakage through any one SG. It is recognized that the leak rate observed during an accident may not necessarily be the same as the leak rate observed during operation (pre-accident) since the loading conditions imposed on the tubes during the accident may be different than that observed during operation.

To clarify that the 0.75 gpm leak rate occurs under accident conditions, the following TS Bases changes are made:

TS Bases B 3.4.13, "RCS Operational Leakage," applicable safety analyses section, Insert 1, third sentence is revised from:

> "The safety analyses for events resulting in steam discharge to the atmosphere, other than SGTR and SLB, assume that primary to secondary LEAKAGE from all SGs is 0.75 gpm (hot conditions)."

to:

> "The safety analyses for events resulting in steam discharge to the atmosphere, other than SGTR and SLB, assume that primary to secondary LEAKAGE from all SGs is 0.75 gpm (hot conditions) under accident conditions."

TS Bases B 3.4.17, "Steam Generator (SG) Tube Integrity," applicable safety analyses section, second paragraph, third sentence is revised from:

> "For other events, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 0.75 gpm."

to:

> "For other events, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 0.75 gpm under accident conditions."

TS Bases B 3.4.17, "Steam Generator (SG) Tube Integrity," limiting condition for operation section, second last paragraph, fourth sentence is revised from:

> "The accident analyses for events other than SGTR and SLB assume that accident induced leakage does not exceed 0.75 gpm total, equally partitioned among the four SGs (approximately 0.19 gpm from each SG)."

to:

> "The accident analyses for events other than SGTR and SLB assume that leakage does not exceed 0.75 gpm total under accident conditions, equally partitioned among the four SGs (approximately 0.19 gpm from each SG)."

The revised TS Bases pages are provided in Enclosure 2.

3. *In the second from last paragraph in the Limiting Condition for Operation section of the Steam Generator Tube Integrity Bases, it is not clear that the proposed wording is consistent with your proposed accident induced leakage performance criterion in Technical Specification 5.5.9.b.2. According to proposed Technical Specification 5.5.9.b.2, the primary to secondary leakage during a SLB is limited to 10.5 gpm (which the staff understands to be your current design and licensing basis); however, no more than 1 gpm can come from sources that have not been specifically exempted from this limit by the NRC (i.e., those from the sources in proposed Technical Specifications 5.5.9.c.1, 5.5.9.c.2, and 5.5.9.c.3). As currently written, your Bases appear to imply that you are allowed to experience 11.5 gpm during a SLB. Please discuss your plans to address this apparent inconsistency.*

PG&E Response:

The primary to secondary leakage during a SLB is limited to 10.5 gpm with no more than 1 gpm coming from sources that were not specifically exempted by the NRC from the 1 gpm limit.

To provide clarification in the TS Bases, the TS Bases B 3.4.17, "Steam Generator (SG) Tube Integrity," limiting condition for operation section, second to last paragraph, third sentence is revised from:

> "For the faulted SG in the SLB event, 10.5 gpm is the accident induced leakage limit for specific sources (as approved by the NRC), and 1 gpm is the accident induced leakage limit for all other sources (i.e., those not specifically exempted by the NRC)."

to:

> "For the faulted SG in the SLB event, 10.5 gpm is the accident induced leakage limit, of which no more than 1 gpm can come from sources not specifically exempted by the NRC from this 1 gpm limit."

The revised TS Bases page is provided in Enclosure 2.

Changes to Technical Specification Bases Pages

Technical Specification Bases Inserts

Insert 1

Safety analyses for design basis events that model primary to secondary LEAKAGE result in steam discharge to the atmosphere. The safety analysis for the SLB event assumes that primary to secondary LEAKAGE is 10.5 gpm (room temperature conditions) from the faulted SG or increases to 10.5 gpm as a result of accident induced conditions, and 0.1 gpm (room temperature conditions) from each intact SG. The safety analyses for events resulting in steam discharge to the atmosphere, other than SGTR and SLB, assume that primary to secondary LEAKAGE from all SGs is 0.75 gpm (hot conditions) under accident conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the SLB safety analysis for the faulted SG.

Insert 2

The SLB is more limiting for site radiation releases for events other than SGTR. The safety analysis for the SLB accident assumes 10.5 gpm primary to secondary LEAKAGE is through the faulted SG. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., small fraction of these limits).

Insert 3

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 7). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

Insert 4

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

Insert 5

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

BASES

APPLICABLE SAFETY ANALYSES	The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a total primary to secondary LEAKAGE rate of 1 gpm from the intact SGs plus the leakage rate associated with a double-ended rupture of a single tube. The SGTR radiological dose analysis assumes loss of off-site power at the time of reactor trip with no subsequent condenser cooling available. The SG PORV for the SG that has sustained the tube rupture is assumed to fail open for 30 minutes, at which time the operator closes the block valve to the PORV. The SGTR radiological dose analysis assumes the contaminated secondary fluid is released briefly to the atmosphere from all the PORVs following reactor trip, is released from the ruptured SG PORV for 30 minutes, is released from the intact SG PORVs during the cooldown, and is released from all PORVs following cooldown until termination of the event. The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) For the SLB event, the primary to secondary LEAKAGE is 10.5 gpm from the faulted SG or is assumed to increase to 10.5 gpm as a result of accident induced conditions, and 0.1 gpm from each intact SG. For other events, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 0.75 gpm under accident conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits). Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program. During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity.

(continued)

BASES

LCO (continued)	Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).
	The accident induced leakage performance criterion ensures (a) that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions, and (b) that the primary to secondary LEAKAGE will not exceed 1 gpm per SG (except for specific types of degradation at specific locations where the NRC has approved greater accident induced leakage) to ensure that the potential for induced leakage during severe accidents will be maintained at a level that will not increase risk. The accident analysis for the SLB event assumes that accident induced leakage does not exceed 10.5 gpm in the faulted SG and 0.1 gpm in each intact SG. For the faulted SG in the SLB event, 10.5 gpm is the accident induced leakage limit, of which no more than 1 gpm can come from sources not specifically exempted by the NRC from this 1 gpm limit. The accident analyses for events other than SGTR and SLB assume that leakage does not exceed 0.75 gpm total under accident conditions, equally partitioned among the four SGs (approximately 0.19 gpm from each SG). The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.
	The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.13, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.
APPLICABILITY	Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.
	RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

(continued)