

# CATEGORY 1

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WOMACK, L.F.      Pacific Gas & Electric Co.  
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SUBJECT: Forwards supplemental info re model 51 SG limited tube support plate analysis for dented or packed tube-to-tube support plate crevices, based on discussions with NRC in March 1999.

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**Pacific Gas and  
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June 8, 1999

PG&E Letter DCL-99-079

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

Supplemental Information - Model 51 Steam Generator Limited Tube Support Plate  
Analysis for Dented or Packed Tube-to-Tube Support Plate Crevices

Dear Commissioners and Staff:

On March 5, 1999, PG&E approved a licensing basis impact evaluation (LBIE) that credited seismic damping values allowed by Regulatory Guide (RG) 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," dated October 1973. RG 1.61 was previously approved in Safety Evaluation Report Supplement No. 7, dated May 26, 1978, and Supplement No. 8, dated November 15, 1978, Section 3.9.3.2, "General Methods of Analysis Description of Methods," for seismic evaluation at Diablo Canyon Power Plant.

The RG 1.61 damping values were used in calculations performed in support of Westinghouse WCAP-14707 (proprietary), and WCAP-14708 (nonproprietary), "Model 51 Steam Generator Limited Tube Support Plate Analysis for Dented or Packed Tube-to-Tube Support Plate Crevices," originally submitted by PG&E Letter DCL-96-206, dated October 4, 1996. Westinghouse WCAP-14707 and WCAP-14708, Revision 1, were transmitted by PG&E Letter DCL-97-104, dated May 30, 1997. PG&E provided information in response to NRC's Request for Additional Information regarding WCAP-14707 and 14708, via PG&E Letter DCL-98-025, dated February 23, 1998, DCL-98-164, dated November 24, 1998, and DCL-99-054, dated April 13, 1999.

Based on discussions with the NRC in March 1999, PG&E is providing the approved Final Safety Analysis Report Update and LBIE to the NRC for information.

Sincerely,

Lawrence F. Womack

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PDR ADDCK. 05000275  
P PDR

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A001



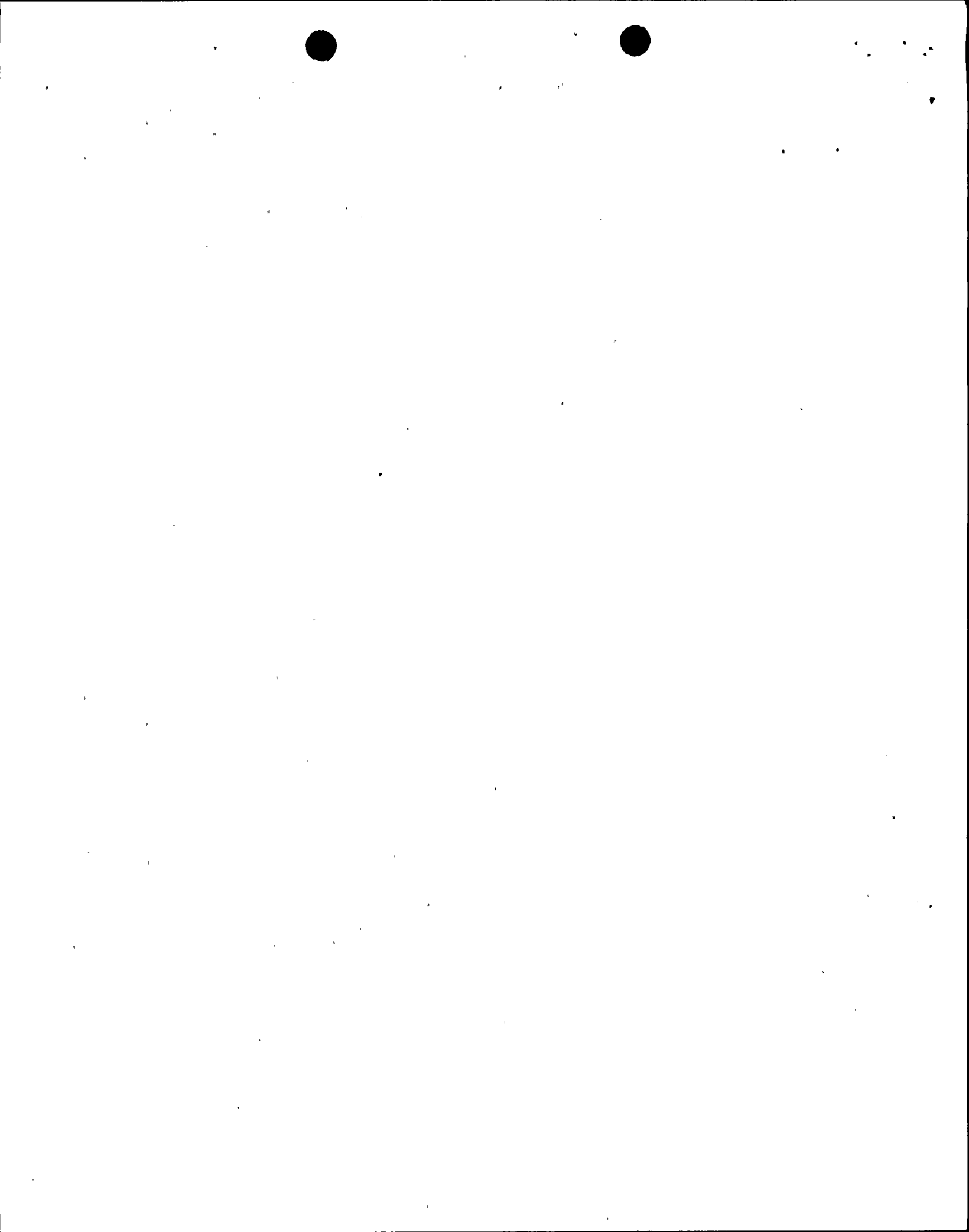
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June 8, 1999  
Page 2 of 2

PG&E Letter DCL-99-079

cc: Steven D. Bloom  
Ellis W. Merschoff  
David L. Proulx  
Diablo Distribution (w/o enc)

Enclosures

DDM/469/N0001999



## FSAR Update Change Request

**1 - FSAR UPDATE REFERENCE (Use Current Living FSAR Update)**

Section(s): 5.5.2.3.4 and 3.7.1.4	Page(s): 5.5-12, -45; 3.7-3, -4, -41, -42
Table(s):	Figure(s):

**2 - DESCRIPTION OF CHANGE**

The proposed change incorporates discussion regarding the locked condition of the SG tubes due to corrosion products having closed the gaps between the SG tubes and tube support plates. References to the analysis that justifies the acceptability of the condition are also included in the proposed change.

**3 - REVISED FSAR UPDATE**

- Attach a mark-up of affected pages from the current Living FSAR Update. Clearly show the proposed changes, additions, and deletions to the text, table(s), figure(s), and appendix(ces).
- Make the change detail consistent with the detail level used in the current FSAR Update. Applicable content guidance is provided in Regulatory Guide 1.70, Rev. 1.

**4 - JUSTIFICATION/REFERENCE**

The justification and references associated for this change are contained in the attached LBIE.

**5 - INITIATOR - I have completed items 1-4 above in accordance with the instructions on page 2.**

Initiator Lee Goyette	Date 3/5/99
Work Group/Organization: NPEQ	

**6 - REVIEWER**

Print Last Name	Signature	Work Group/Org.	Date

**7 - APPROVER**

Print Last Name <b>SHOULDERS</b>	Signature	Date <b>3/5/99</b>
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**FOR GENERIC LICENSING USE ONLY**

Evaluated & Accepted:	Date	Change Number:
Incorporated in Living FSAR Update:	Date:	
Living FSAR Update Verified:	Date:	





**Initiator**

**Block 1** - Using the current Living FSAR Update, identify the section(s), page(s), table(s), and/or figure(s) to be changed.

**Block 2** - Provide a brief description of the proposed change.

**Block 3** - Self explanatory.

**Block 4** - Search for and revise any other parts of the FSAR Update affected by the proposed change. Provide justification and references for the proposed change as follows:

If the proposed change ...	Then ...
Results from a change in the DCP facility or procedures	<ul style="list-style-type: none"> <li>• Record the design change document number (e.g., DCP, MMP) or the procedure number.</li> <li>• Attach a copy of the LBIE Screen/LBIE (TS3.ID2). Include documentation of PSRC review and approval for LBIEs.</li> </ul>
Is the result of a license amendment, regulatory commitment, or NRC requirement	<ul style="list-style-type: none"> <li>• Record the identification of the source (License Amendment number, letter number, etc.)</li> <li>• Attach a copy of relevant sections of the documents justifying the proposed change.</li> </ul>
Is editorial, such as correcting typos or clarifying the meaning of an existing statement	<ul style="list-style-type: none"> <li>• State that the proposed change meets the conditions of an editorial correction.</li> <li>• No LBIE Screen/LBIE is needed.</li> </ul>
Is not categorized by any of the above	<ul style="list-style-type: none"> <li>• Prepare and attach a copy of an LBIE Screen/LBIE. Include PSRC review and Plant Manager approval documentation for LBIEs.</li> </ul>

**Block 5** -

- Print name, date and your work group or organization.
- Obtain reviews from other work groups if deemed necessary.
- Obtain your supervisor's approval.
- Submit completed request to Generic Licensing. Include signed LBIE Screen and PSRC-reviewed and Plant Manager approved LBIE, when required to support change requests.

**Reviewer(s)**

**Block 6** - When other reviews are necessary, print name, sign, date, and identify work group/org.

**Approver**

**Block 7** - Initiator's supervisor print name, sign, and date to indicate approval of change.

**FOR GL USE ONLY**

- Verify change request is complete, including necessary reviews and approvals. If not, return to initiator
- Assure change request's content is consistent with the level of detail in the current FSAR Update and conforms to Regulatory Guide 1.70, Revision 1 guidance.
- GL engineer sign, date and assign a change number when change is acceptable (applicable LBIE Screen/LBIE is included with change and PCD/procedure impact determined).
- GL engineer sign and date to signify log updated and change incorporated into Living FSAR Update.
- Independent GL engineer sign and date to signify Living FSAR Update entries are correct.



#### 5.5.2.3.4 Tube and Tubesheet Stress Analyses

Tube and tubesheet stress analyses of the steam generator are discussed in Reference 4. The calculations confirm that the steam generator tubesheet will withstand the loading (quasi-static rather than shock loading) caused by loss of reactor coolant. With the acceptance of the DCPD leak-before-break analysis by the NRC (Reference 10), dynamic loading conditions resulting from pipe rupture events in the main reactor coolant loop piping no longer have to be considered in the design basis analyses; only the much smaller dynamic loads resulting from RCS branch line breaks have to be considered (see Section 3.6.2.1.1.1). The tube-locking effects associated with the formation of corrosion products in the tube to tube support plate gaps have been evaluated in a supplemental analysis (Reference 11).

#### 5.5.2.3.5 Corrosion

An evaluation was performed to determine the extent of tube wall thinning that can be tolerated under accident conditions. The worst case loading conditions are assumed to be imposed upon uniformly thinned tubes at the most critical location in the steam generator. Under such a postulated design basis accident, vibration is of short enough duration that there is no endurance problem. The results of a study made on Model 51 (0.875-inch nominal diameter, 0.050-inch nominal thickness) tubes under accident loading (discussed in Reference 4) showed that a minimum wall thickness of 0.021 inches would have a maximum faulted condition stress (due to combined LOCA and safe shutdown earthquake loads) that is less than the allowable limit.

The assumed corrosion loss was based on a conservative weight loss rate for Inconel tubing in flowing 650°F primary side reactor coolant fluid. The weight loss, when equated to a thinning rate and projected over a 40-year design objective including appropriate reduction for initial protective film formation, is equivalent to 0.000083-inch thinning. The assumed corrosion loss of 0.003 inches leaves a conservative 0.002917 inches for general corrosion thinning on the secondary side. The steam generator tubes, existing originally at their minimum wall thickness and reduced by a very conservative general corrosion loss, still provide an adequate safety margin.

Steam generator tubing has been successfully tested for its compatibility with primary and secondary coolants. Testing to investigate the susceptibility of heat exchanger construction materials to stress corrosion caustic and chloride aqueous solutions indicated that Inconel 600 has resisted general corrosion in severe operating water conditions. Many years of successful reactor operation have shown the same low general corrosion rates indicated by the laboratory tests.

Recent operating experience, however, has revealed areas on the secondary side where localized corrosion rates were significantly greater than the low general corrosion rates. Denting, intergranular corrosion, and tube wall thinning were experienced in localized areas,



DCPP UNITS 1 & 2 FSAR UPDATE

11. Tube Structural Evaluation for Diablo Canyon Units 1 and 2 Under Packed Conditions, NSD-E-SGDA-98-334 / SG-98-10-003, Westinghouse Electric Company, November 1998.



Figure 3.7-3 (2 percent damping) and in Figure 3.7-4 (5 percent damping). These spectra are calculated at period intervals of 0.01 seconds, which adequately define the spectra.

For the HE evaluation of containment structure, auxiliary building, turbine building, and intake structure, the horizontal input motions are reduced from free-field motions to account for the presence of the structures that have large foundations. These reduced inputs have been derived by spatial averaging of acceleration across the foundations of each structure by the Tau filtering procedure (Reference 12). The resulting horizontal response spectra for these structures are shown in Figures 3.7-4A through 3.7-4F.

For HE evaluation of outdoor water storage tanks and smaller structures, the horizontal design response spectra are the free-field horizontal response spectra. HE vertical design response spectra are the free-field vertical response spectra. For design purposes, the Newmark spectra are used, or alternately the Blume spectra are used, with adjustment in certain frequency ranges as necessary so that they do not fall below the corresponding Newmark spectra.

Acceleration time-histories used in the analysis of the containment and intake structures, auxiliary building, and turbine building are shown in Figures 3.7-4G through 3.7-4M. Comparison of the response spectrum computed from each time-history with the corresponding design response spectrum for 7 percent damping is shown in Figures 3.7-4N through 3.7-4T.

### 3.7.1.3 Critical Damping Values

The specific percentages of critical damping used for Design Class I SSCs, and the Design Class II turbine building and intake structure are listed in the following table:

<u>Type of Structure</u>	<u>% of Critical Damping</u>		
	<u>DE</u>	<u>DDE</u>	<u>HE</u>
Containment structures and all internal concrete structures	2.0	5.0	7.0
Other conventionally reinforced concrete structures above ground, such as shear walls or rigid frames	5.0	5.0	7.0
Welded structural steel assemblies	1.0	1.0	4.0
Bolted or riveted steel assemblies	2.0	2.0	7.0
Mechanical components (PG&E purchased)	2.0	2.0	4.0
Vital piping systems (except reactor coolant loop) <sup>(a)</sup>	0.5	0.5	3.0 <sup>(b)</sup>

<sup>(a)</sup> ASME Code Case N-411 damping may be used provided it is applied to all earthquake cases and used in response spectrum modal superposition analysis. When used, pipe displacements are checked for adequacy of clearances and pipe mounted equipment accelerations are verified against project qualification criteria.

For equipment and components modeled inline, damping should be consistent with RG 1.61; a composite damping value may be used for the analysis of these piping systems.

A log of calculations is kept that indicates which calculations have used Code Case N-411 damping. Request for NRC approval for the use of ASME Code Case N-411 was made in letter DCL-86-009, dated January 22, 1986. NRC approval was granted by letter on April 7, 1986.





Reactor coolant loop <sup>(c) (a)</sup>	0.5	1.0	4.0
Foundation rocking (containment structure only) <sup>(d)</sup>	5.0	5.0	NA <sup>(e)</sup>

### 3.7.1.4 Bases for Site-Dependent Analysis

Site conditions used to develop the shape of site seismic design response spectra are described in Section 2.5.2.

### 3.7.1.5 Soil-Supported Design Class I Structures

All Design Class I plant structures are founded on rock or on concrete fill.

### 3.7.1.6 Soil-Structure Interaction

Soil-structure interaction effects are considered as described in Section 3.7.2.1.7.

### 3.7.1.7 Hosgri Evaluation

The criteria and methods used to review the major structures for response to the postulated 7.5M HE are discussed in this chapter. A comparison of the DE and the DDE criteria with the HE evaluation criteria is given in Table 3.7-1 for the containment and auxiliary building, Tables 3.7-1A for the turbine building, 3.7-1B for the intake structure, and 3.7-1C for the outdoor water storage tanks, respectively.

## 3.7.2 SEISMIC SYSTEM ANALYSIS

In accordance with Revision 1 to RG 1.70, paragraphs under the headings below Seismic Analysis Methods and Description of Seismic Analyses, apply to all seismic analysis performed, i.e., both seismic system analysis and seismic subsystem analysis. Paragraphs under subsequent headings in this section provide discussion of specific topics applicable to seismic system analysis. Discussion of specific topics applicable to seismic subsystem analysis

- Request for NRC approval for the use of ASME Code Case N-411 was made in letter DCL-86-009, dated January 22, 1986. NRC approval was granted by letter on April 7, 1986.
- (b) Two percent of critical damping is used for piping less than or equal to 12 inches in diameter.
  - (c) Although a damping value of 1 percent is used for the DDE analysis of the reactor coolant loop (RCL), damping values of greater than 4 percent have been measured experimentally for the RCL in full-size power plants<sup>(8)</sup>. These testing programs have been reviewed and approved by the NRC. The damping values recommended in Reg Guide 1.61 are acceptable for use in analysis of mechanical equipment and systems<sup>(24, 23, 26)</sup>.
  - (d) Five percent of critical damping is used for structures founded on rock for the purpose of computing the response in the rocking mode, and 7 percent of critical damping is used for the purpose of computing the response in the translation mode.
  - (e) Analysis utilizes fixed base.



DCPP UNITS 1 & 2 FSAR UPDATE

12. Supplement No. 5 to the Safety Evaluation of the Diablo Canyon Nuclear Power Station, Units 1 and 2, Nuclear Regulatory Commission, Division of Reactor Licensing, Washington, DC, September 1976.
13. "Dynamics of Fixed-Base Liquid Storage Tanks," Veletsos, A.S. and T.Y. Yang; Proceedings of U.S.-Japan Seminar on Earthquake Engineering Research with Emphasis on Lifeline Systems, Tokyo, November 1976.
14. Westinghouse 1981 ECCS Evaluation Model Using the BASH Code, WCAP-10266-P-A, Rev. 2, March 1987.
15. Seismic Evaluation for Postulated 7.5M Hosgri Earthquake, DCPP Units 1&2, PG&E.
16. PG&E Design Change Package N-47546.
17. PG&E Letter to the NRC, DCL-92-198 (LER 1-92-015).
18. Phase I Final Report - Design Verification Program, Diablo Canyon Power Plant, Revision 14, transmitted via letter dated October 14, 1983, J. O. Schuyler (PG&E) to D. G. Eisenhower (NRC).
19. Final Report of the Diablo Canyon Long Term Seismic Program, July 1988, PG&E.
20. Addendum to the 1988 Final Report of the Diablo Canyon Long Term Seismic Program, February 1991, PG&E.
21. NUREG-0675, Supplement Number 34, Safety Evaluation Report Related to the Operation of Diablo Canyon Nuclear Power Plant, Units 1 and 2, NRC, June 1991.
22. NRC letter to PG&E, "Transmittal of Safety Evaluation Closing Out Diablo Canyon Long-Term Seismic Program," April 17, 1992.
23. PG&E letter to the NRC, "Long Term Seismic Program - Future Plant Modifications," DCL-91-178, July 16, 1991.
24. Supplement No. 7 to the Safety Evaluation of the Diablo Canyon Nuclear Power Station, Units 1 and 2, Nuclear Regulatory Commission, Division of Reactor Licensing, Washington, DC, May 1978.
25. Supplement No. 8 to the Safety Evaluation of the Diablo Canyon Nuclear Power Station, Units 1 and 2, Nuclear Regulatory Commission, Division of Reactor Licensing, Washington, DC, November 1978.



26. Damping Values for Seismic Design of Nuclear Power Plants, Regulatory Guide 1.61, USAEC, October 1973.



## NUCLEAR POWER GENERATION

TS3.ID2

## ATTACHMENT 8.1

TITLE: LICENSING BASIS IMPACT EVALUATION SCREEN *LBIE No. 99-036*

## 1 - BASIC INFORMATION

REFERENCE DOCUMENT NO.FSAR	DOC. REV. NO.13
REFERENCE DOCUMENT TITLE: FSAR Sections 5.5.2.3.4 and 3.7.1.3	
SPONSORING ORGANIZATION: Design Engineering	SPONSOR (PRINT LAST NAME): Goyette

## 2 - DESCRIPTION

During the 1996 review of the Westinghouse generic limited tube support plate displacement analysis for dented or packed crevices (WCAP-14707), PG&E determined that the locked-tube condition existing in the Diablo Canyon (DCPP) Units 1 & 2 Steam Generators (SG) may affect assumptions in the stress analysis of the tubes and tube support plates (TSP). It was also noted that tube locking introduces a thermal loading condition (constrained thermal expansion) that was not considered in the design analysis. Accordingly, PG&E initiated action to evaluate the effect of this aging-related condition and to demonstrate the continued operability of the steam generators.

One of these actions is the structural evaluation of locked tubes and TSP according to the ASME Code (Reference 1). This assessment is performed to demonstrate the continued operability of the SGs in the present condition. The operability analysis is intended to supplement but not supersede the original design basis analysis; it is to be referenced as such in the FSAR.

This operability analysis uses the damping values recommended in Reg Guide 1.61 (Reference 2) as approved for use at DCPP (Reference 3, 4). As NRC approval of these recommended damping values is implied, but not clearly reflected in FSAR section 3.7.1.3 "Critical Damping Values" (Reference 5), the FSAR entry must be clarified.

## 3 - SCREENING FOR CHANGES TO THE OPERATING LICENSE

3.1	SECTION 1: <u>Screen for Changes to the Operating License</u> Does this activity or CTE involve a change to the Facility Operating License (OL), including OL Attachments (Technical Specifications, Environmental Protection Plan and Antitrust Conditions)? If "yes," contact Regulatory Services. A license amendment must be received from the NRC before an activity or CTE impacting the OL is implemented.	<input type="checkbox"/> Y	<input checked="" type="checkbox"/> N
3.2	SECTION 2: <u>Screen for Regulatory Commitments and Obligations</u>		
a)	Does this activity or CTE impact a regulatory commitment or obligation contained in the PCD? If "yes," process the change in accordance with IDAP XI4.ID2.	<input type="checkbox"/> Y	<input checked="" type="checkbox"/> N
b)	According to IDAP XI4.ID2, does the activity or CTE impact an obligation that requires prior NRC approval? If "yes," contact Regulatory Services. NRC approval must be received before an activity or CTE impacting the obligation is approved.	<input type="checkbox"/> Y	<input checked="" type="checkbox"/> N





TS3.ID2  
ATTACHMENT 8.1

**TITLE: LICENSING BASIS IMPACT EVALUATION SCREEN**

**4 - SCREENING FOR DETERMINING THE NEED FOR A LBIE**

<b>4.1</b>	<b>SECTION 1: <u>10 CFR 50.59 Safety Evaluation Screen</u></b>		
	a) Does this activity or CTE involve a change to the facility design, function, or method of performing the function as described in the SAR, including text, tables, and figures?	<input checked="" type="checkbox"/> Y	<input type="checkbox"/> N
	b) Does this activity or CTE involve a change to procedures, system operation, or administrative control over plant activities as described in the SAR, including text, tables, and figures?	<input type="checkbox"/> Y	<input checked="" type="checkbox"/> N
	c) Does this activity or CTE result in a test, experiment, condition, or configuration that might affect safe operation of the plant but was not anticipated, described, or evaluated in the SAR?	<input checked="" type="checkbox"/> Y	<input type="checkbox"/> N
	d) Does this activity or CTE rely on a vendor safety evaluation which has not been reviewed by the PSRC?	<input type="checkbox"/> Y	<input checked="" type="checkbox"/> N
A "yes" response to any of the Section 1 questions requires Section 1 of Form 69-10431 (LBIE) to be completed.			
<b>4.2</b>	<b>SECTION 2: <u>Fire Protection Program Screen</u></b>		
	Does the activity or CTE involve a change to the Fire Protection Program as described in FSAR Update Chapter 9.5?	<input type="checkbox"/> Y	<input checked="" type="checkbox"/> N
A "yes" response to Section 2 questions requires Section 2 of Form 69-10431 (LBIE) to be completed.			
<b>4.3</b>	<b>SECTION 3: <u>Quality Assurance Program Screen</u></b>		
	Does the activity or CTE involve a change to the Quality Assurance Program as described in FSAR Update Chapter 17?	<input type="checkbox"/> Y	<input checked="" type="checkbox"/> N
A "yes" response to Section 3 requires Section 3 of Form 69-10431 (LBIE) to be completed.			
<b>4.4</b>	<b>SECTION 4: <u>Environmental Protection Screen</u></b>		
	a) Does this activity or CTE have a potential effect on the environment on the basis of Appendix 7.3? If "no," skip the next question and signature.	<input type="checkbox"/> Y	<input checked="" type="checkbox"/> N
	b) If "yes," does the activity or CTE result in a change to the Environmental Protection Plan or create a situation that may be adverse to the environment?	<input type="checkbox"/> Y	<input type="checkbox"/> N
A "yes" response to question b) requires Section 4 of Form 69-10431 (LBIE) to be completed.			
<b>ENVIRONMENTAL REVIEWER SIGNATURE</b>		<b>DATE</b>	<b>PRINT LAST NAME</b>
<b>4.5</b>	<b>SECTION 5: <u>Emergency Plan Screen</u></b>		
	a) Does the Emergency Plan (EP) require review on the basis of Appendix 7.4? If "no," skip the next question and signature.	<input type="checkbox"/> Y	<input checked="" type="checkbox"/> N
	b) If "yes," does the activity or CTE result in a change to the EP?	<input type="checkbox"/> Y	<input type="checkbox"/> N
A "yes" response to question b) requires Section 5 of Form 69-10431 (LBIE) to be completed.			
<b>EMERGENCY PLAN REVIEWER SIGNATURE</b>		<b>DATE</b>	<b>PRINT LAST NAME</b>
<b>4.6</b>	<b>SECTION 6: <u>Security Plans Screen</u></b>		
	a) Do any of the security plans (PSP, SCP, STOP) require review on the basis of Appendix 7.5? If "no," skip the next question and signature.	<input type="checkbox"/> Y	<input checked="" type="checkbox"/> N
	b) If "yes", does the activity or CTE result in a change to a security plan? If so, which plan(s)	<input type="checkbox"/> Y	<input type="checkbox"/> N
A "yes" response to question b) requires Section 6 of Form 69-10431 (LBIE) to be completed.			
<b>SECURITY PLAN REVIEWER</b>		<b>DATE</b>	<b>PRINT LAST NAME</b>



TS3.ID2  
ATTACHMENT 8.1

**TITLE: LICENSING BASIS IMPACT EVALUATION SCREEN**

**5. IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES REVIEW**

(The proposed ITS and Bases are located in EDMS under NPG Library/Regulatory Documents/Improved Technical Specifications).

Does the activity or CTE affect the proposed ITS or the proposed ITS Bases including references listed in the Bases?  Y  N

If "yes," complete the ITS and Bases Change Request Form in TP TA-9802 and forward to Regulatory Services.

**6. REMARKS**

4.1 a) The closing of design gaps between the SG tubes and tube support plates may be considered a *de facto* design change in that it affects the SG design analysis.

4.1 c) The closing of design gaps between the SG tubes and tube support plates results in an unanticipated structural configuration.

**7. REFERENCES/ATTACHMENTS**

Reference:

1. Tube Structural Evaluation for Diablo Canyon Units 1 and 2 Under Packed Conditions, Calculation NSD-E-SGDA-98-334 / SG-98-10-003, Westinghouse Electric Company, November, 1998.
2. Damping Values for Seismic Design of Nuclear Power Plants, Regulatory Guide 1.61, USAEC, October 1973.
3. Supplement No. 7 to the Safety Evaluation of the Diablo Canyon Nuclear Power Station, Units 1 and 2, Nuclear Regulatory Commission, Division of Reactor Licensing, Washington, DC, May 1978.
4. Supplement No. 8 to the Safety Evaluation of the Diablo Canyon Nuclear Power Station, Units 1 and 2, Nuclear Regulatory Commission, Division of Reactor Licensing, Washington, DC, November 1978.
5. FSAR section 3.7.1.3, "Critical Damping Values"

**8. SCREEN CONCLUSIONS**

Based upon the above criteria, I have determined that an LBIE:  is  is not required.

PREPARER SIGNATURE <i>J. Goyette</i>	DATE <i>3/5/99</i>	PRINT LAST NAME Goyette
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Based upon my independent technical review, I concur with the above conclusion:

INDEPENDENT TECHNICAL REVIEWER SIGNATURE <i>[Signature]</i>	DATE <i>3-5-99</i>	PRINT LAST NAME EXNER
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NUCLEAR POWER GENERATION  
TS3.ID2  
ATTACHMENT 8.2

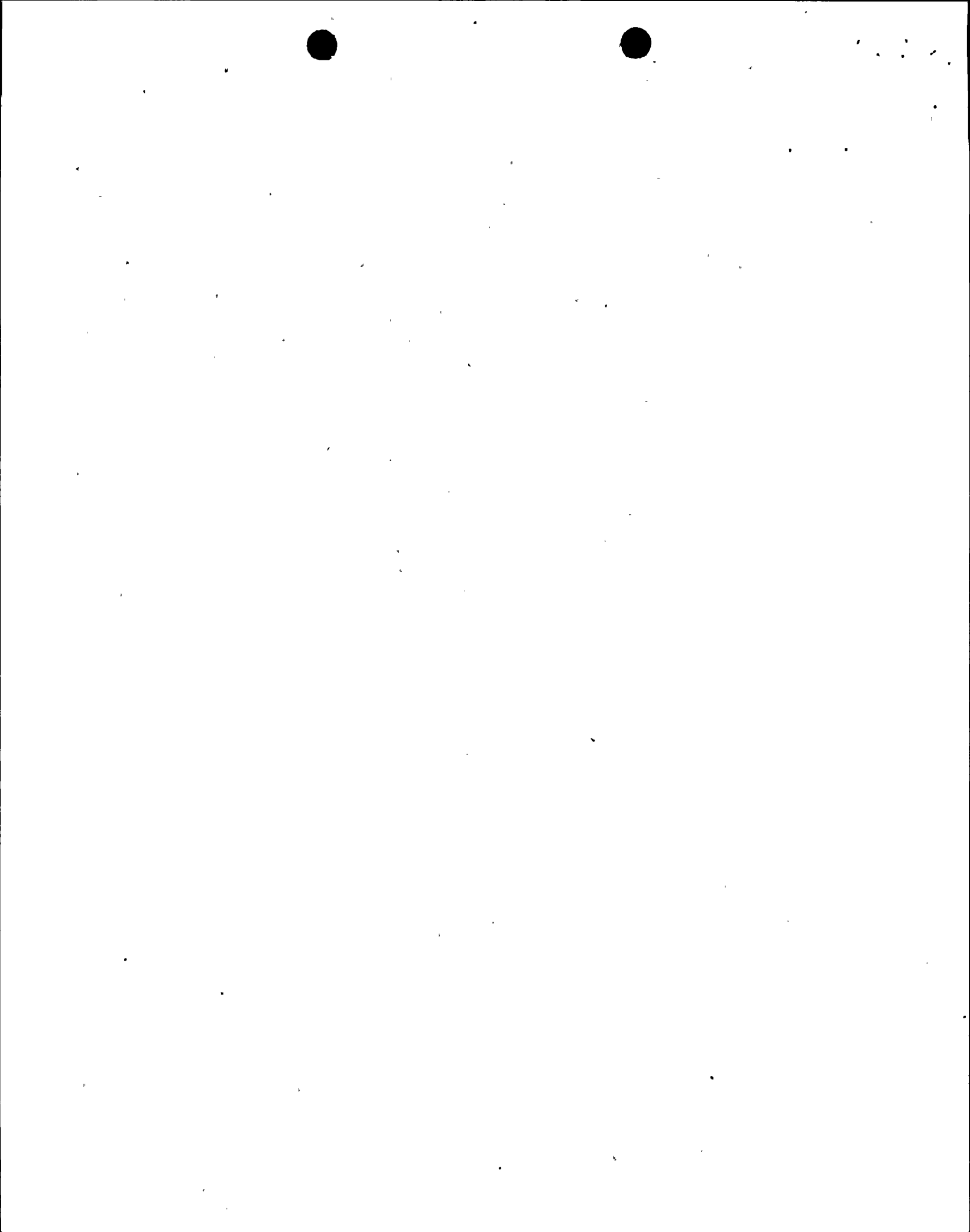
TITLE: LICENSING BASIS IMPACT EVALUATION

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**Section 0: General Information and Summary**

LBIE TITLE: Steam Generator Tubes Locked in Tube Support Plates		LBIE NUMBER/REV. NO. 99-036
REFERENCE DOCUMENT NO.: FSAR	DOC. REV. NO.: 13	
REFERENCE DOCUMENT TITLE: FSAR Sections 5.5.2.3.4 and 3.7.1.3		
SPONSORING ORGANIZATION: Design Engineering	SPONSOR PRINT LAST NAME Goyette	
PSRC MEETING NO. 99-026	DATE: 3/5/99	APPROVAL RECOMMENDED: <input checked="" type="checkbox"/> Y <input type="checkbox"/> N
PLANT MANAGER APPROVAL: <i>[Signature]</i>		DATE: 3/5/99

**Safety Evaluation Summary**



TS3.ID2  
ATTACHMENT 8.2

TITLE: LICENSING BASIS IMPACT EVALUATION

Change Description

Normal corrosion processes associated with operation of the DCPD steam generators (SG) have caused the as-designed gaps between the SG tubes and the tube support plates (TSP) to become filled with corrosion products. This creates a fixed boundary condition at the tube-TSP interface that was considered neither in the original analysis, nor in the original design. Recent analysis (Reference 1) of the SGs in this locked-tube condition demonstrate compliance with ASME Code requirements, hence acceptability for continued operation. The FSAR is to be updated to reference this analysis.

The operability analysis uses the damping values recommended in Reg Guide 1.61 (Reference 2) as approved for use at DCPD (Reference 3, 4). As NRC approval of these recommended damping values is implied, but not clearly reflected in FSAR section 3.7.1.3 "Critical Damping Values" (Reference 5), the FSAR entry must be clarified.

Licensing Basis Impacts

1. FSAR section 3.7.1.3, "Critical Damping Values"
2. FSAR section 5.5.2.3.4, "Tube and Tubesheet Stress Analysis"

Narrative Summary

This LBIE addresses FSAR changes only: 1) Including reference to the operability analysis for the locked-tube condition, and 2) Clarification of the acceptable use of Reg Guide 1.61 recommended damping values. This LBIE is somewhat out of the ordinary in that the physical change that makes necessary the first FSAR change above has taken place as a result of normal equipment operation—it is not a change to be implemented by the modification process. There is no physical mod to be implemented.

While investigating NRC Information Notice 96-09 "Damage in Foreign Steam Generator Internals," PG&E recognized that the design analysis of record for the DCPD SGs does not properly reflect the fixed boundary conditions caused by the closing of the tube-TSP crevices. PG&E initiated action to evaluate the effects of tube-locking, and engaged Westinghouse Electric Company to perform analysis to demonstrate Code compliance of the operating steam generators. The analysis is to be referenced in section 5.5.2.3.4 "Tube and Tubesheet Stress Analysis" as a supplement to the (original) analysis of record.

FSAR section 3.7.1.3 "Critical Damping Values" contains in note "c" an ambiguous reference to acceptable use of the Reg Guide 1.61 damping values rather than the tabular damping values (i.e., 0.5% DE, 1.0% DDE, 4.0% Hosgri) for the reactor coolant loop. SSER 7 (Reference 3, page 3-48) states in part "The damping values recommended by Reg Guide 1.61 constitute our current criteria and have been acceptable on all applications for several years. Therefore, they are acceptable for use in the seismic reevaluation." referring to the Hosgri analysis. SSER 7 continues with a statement concerning NRC acceptance of the full-size plant tests (Reference 6) and a request that PG&E demonstrate the similarity of DCPD SGs to those of Indian Point 2. This demonstration was accepted in SSER 8 (Reference 4). Thus, the note in FSAR section 3.7.1.3 will be expanded to clearly indicate the acceptability of Reg Guide 1.61 damping values (i.e., 2% DE, 4% DDE, 4% Hosgri).

Conclusion

Both proposed changes to the FSAR are acceptable.





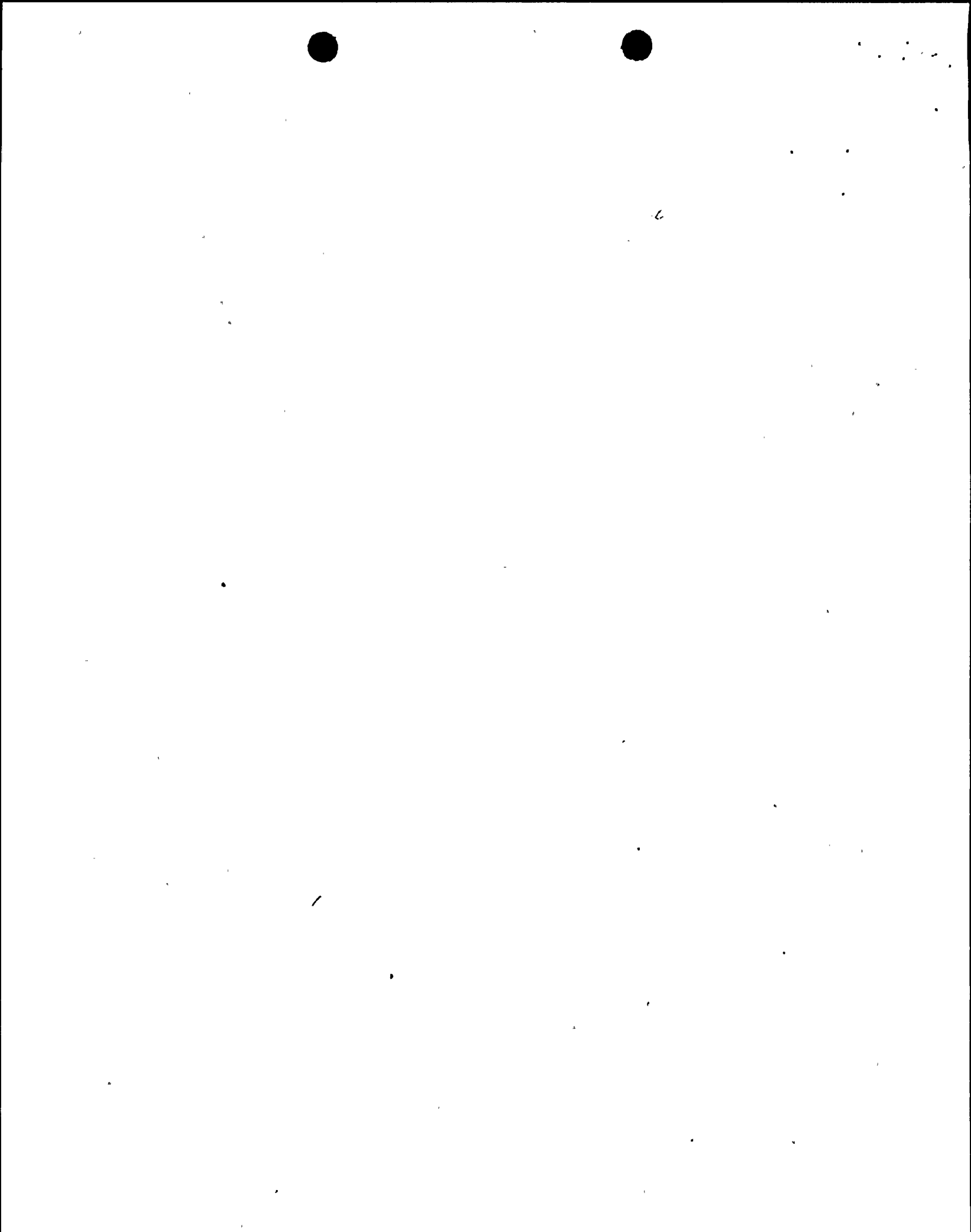
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ATTACHMENT 8.2

TITLE: LICENSING BASIS IMPACT EVALUATION

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**References**

1. "Tube Structural Evaluation for Diablo Canyon Units 1 and 2 Under Packed Conditions," Calculation NSD-E-SGDA-98-334 / SG-98-10-003, Westinghouse Electric Company, November 1998.
2. Damping Values for Seismic Design of Nuclear Power Plants, Regulatory Guide 1.61, USNRC, October 1973.
3. Supplement No. 7 to the Safety Evaluation of the Diablo Canyon Nuclear Power Station, Units 1 and 2, Nuclear Regulatory Commission, Division of Reactor Licensing, Washington, DC, May 1978.
4. Supplement No. 8 to the Safety Evaluation of the Diablo Canyon Nuclear Power Station, Units 1 and 2, Nuclear Regulatory Commission, Division of Reactor Licensing, Washington, DC, November 1978.
5. "Critical Damping Values," FSAR section 3.7.1.3
6. "Damping Values of Nuclear Power Plant Components," WCAP-7921-AR, Westinghouse Electric Corporation, November 1972.



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**TITLE: LICENSING BASIS IMPACT EVALUATION**

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**Section 1: 10 CFR 50.59 Safety Evaluation**

- |    |   | Yes                      | No                                  |
|----|---|--------------------------|-------------------------------------|
| 1. | <p>May the probability of occurrence of an accident previously evaluated in the SAR be increased?</p> <p>Justification:<br/>The analyzed accidents relevant to this topic are steam generator tube rupture (SGTR, FSAR Chapter 15.4.3) and Steam Line Break (MSLB, FSAR Chapter 15.4.2). At the most basic level, the SGTR event signifies the failure of pressure boundary material, which may be caused by gross overstress or fatigue. MSLB loadings are included in the ASME Code evaluation. The ASME Code acceptance criteria is based on specific design margin(s) to failure; acceptability under the Code demonstrates a specific degree of confidence against material failure. Thus, the design analysis of record explicitly implies a certain probability of failure.</p> <p>The operability analysis (Reference 1) demonstrates continued Code acceptability in the locked-tube condition. Thus there is no increase in the probability of occurrence of SGTR nor MSLB.</p> | <input type="checkbox"/> | <input checked="" type="checkbox"/> |
| 2. | <p>May the consequences of an accident previously evaluated in the SAR be increased?</p> <p>Justification:<br/>The consequences of SGTR and MSLB have already been evaluated. The parameters that could potentially affect the consequences of a SGTR include the initial RCS activity, the method of release from the SG to the atmosphere, and the time required to fill the SG. The proposed change does not affect the RCS activity or the release path. The increased stress on the SG tubes could result in an increase in the number of failed tubes. However the Westinghouse evaluation (Reference 1) demonstrates continued compliance with ASME code and therefore does not change the probability of any more tubes than assumed in the SGTR analysis failing. The MSLB loadings are also included in the ASME Code evaluation and have been found acceptable. Therefore, the proposed change does not result in an increase in the consequences of either accident.</p>      | <input type="checkbox"/> | <input checked="" type="checkbox"/> |
| 3. | <p>May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?</p> <p>Justification:<br/>The equipment affected by the proposed change is the SG tubes. The SG tubes constitute the primary/secondary pressure boundary. The SG pressure boundary is passive equipment, and as such the limiting malfunction is a failure of that pressure boundary. The Westinghouse analysis demonstrates that the tubes remain with the code allowable stresses and assures that the tubes are no more likely to malfunction than previously. Therefore, the probability of occurrence of a malfunction of equipment important to safety is not increased.</p>   | <input type="checkbox"/> | <input checked="" type="checkbox"/> |
| 4. | <p>May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?</p> <p>Justification:</p>   | <input type="checkbox"/> | <input checked="" type="checkbox"/> |

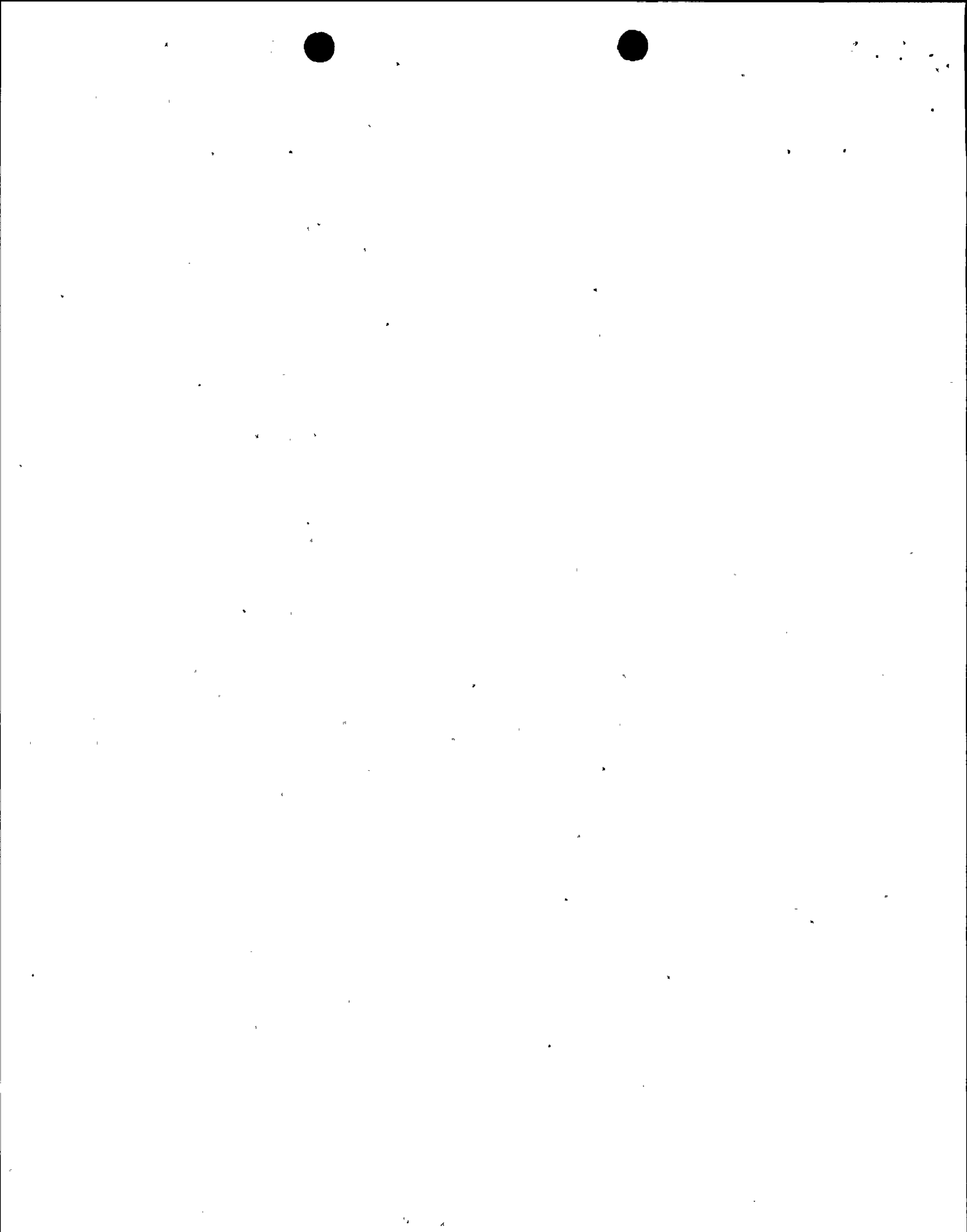


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**TITLE: LICENSING BASIS IMPACT EVALUATION**

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A SGTR is the only event that can occur as a result of the proposed change. The consequences of a SGTR are already evaluated and will not change since the locked-tube condition still satisfies the ASME Code and indicates that no more tubes than assumed will fail during SGTR. The MSLB loadings are included in the ASME Code evaluation, and there would be no additional primary-secondary leakage created by the locked tubes. As stated in Questions 2 and 3, above, failure (i.e., malfunction) of the tube pressure boundary is the limiting case. Therefore, the consequences of a "malfunction" are unchanged.



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**TITLE: LICENSING BASIS IMPACT EVALUATION**

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**Section 1: 10 CFR 50.59 Safety Evaluation (continued)**

Yes No

5. May the possibility of an accident of a different type than any previously evaluated in the SAR be created?  Yes  No  
**Justification:**  
 Pressure boundary failure is the limiting accident associated with SG tubes—there are no other credible failure modes. The failure of additional SG tubes beyond those assumed in the SGTR analysis could potentially be considered a different type of accident; however, the Westinghouse analysis demonstrates that the stress on the tubes is within the ASME Code allowables. Consequently, the tubes are no more likely to fail than pervasively. Therefore, the proposed application of the Westinghouse analysis does not create the possibility of a different type than any previously evaluated.
6. May the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR be created?  Yes  No  
**Justification:**  
 Failure of the SG tube pressure boundary has no bearing on malfunction of other equipment. Failure of SG tubes has already been evaluated as part of the SGTR accident analysis. Therefore, the proposed change does not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated.
7. Is there a reduction in the margin of safety as defined in the basis for any Technical Specification?  Yes  No  
**Justification:**  
 FSAR section 3.7.1.3 "Critical Damping Values" contains in note "c" an ambiguous reference to acceptable use of the Reg Guide 1.61 damping values rather than the tabular damping values (i.e., 0.5% DE, 1.0% DDE, 4.0% Hosgri) for the reactor coolant loop. SSER 7 (Reference 3, page 3-48) states in part "The damping values recommended by Reg Guide 1.61 constitute our current criteria and have been acceptable on all applications for several years. Therefore, they are acceptable for use in the seismic reevaluation." referring to the Hosgri analysis. SSER 7 continues with a statement concerning NRC acceptance of the full-size plant tests (Reference 6) and a request that PG&E demonstrate the similarity of DCPG SGs to those of Indian Point 2. This demonstration was accepted in SSER 8 (Reference 4). Thus, the note in FSAR section 3.7.1.3 will be expanded to clearly indicate the acceptability of Reg Guide 1.61 damping values (i.e., 2% DE, 4% DDE, 4% Hosgri).  
 Satisfaction of ASME Code allowables in the analysis continues to support the margin of safety inherent in the design analysis of record. The analysis has no bearing on the amount of primary to secondary leakage associated with MSLB.





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TITLE: LICENSING BASIS IMPACT EVALUATION

CONCLUSION

Based on the above, the preparer and ITR have determined that an unreviewed safety question  is  is not involved.

If an unreviewed safety question is involved, NRC approval is required prior to implementing the activity.

10 CFR 50.59 Safety Evaluation Signatures		<input type="checkbox"/> N/A
PREPARER SIGNATURE <i>[Signature]</i>	DATE: 3/5/99	PRINT LAST NAME Goyette
ITR SIGNATURE <i>[Signature]</i>	DATE: 3-5-99	PRINT LAST NAME EXNER



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ATTACHMENT 8.2

**TITLE: LICENSING BASIS IMPACT EVALUATION**

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**Section 2: Fire Protection Program Evaluation**

Yes No

Does the change to the Fire Protection Program adversely affect the ability to achieve and maintain safe shutdown in the event of a fire?

Changes which adversely affect the ability to active and maintain safe shutdown in the event of a fire require prior approval of the NRC OL Condition 2.C.(5)b./2.C.(4)(b).

Fire Protection Program Evaluation Signatures		<input checked="" type="checkbox"/> N/A
PREPARER SIGNATURE	DATE:	PRINT LAST NAME
ITR SIGNATURE	DATE:	PRINT LAST NAME

**Section 3: Quality Assurance Program Evaluation**

Yes No

Does the change to the Quality Assurance Program, as described in FSAR Update Chapter 17, reduce the program commitments?

Changes to the Quality Assurance Program, as described in FSAR Update Chapter 17, which reduce program commitments must be submitted to the NRC and receive NRC approval prior to implementation in accordance with 10 CFR 50.54(a)(3).

Quality Assurance Program Evaluation Signatures		<input checked="" type="checkbox"/> N/A
PREPARER SIGNATURE	DATE:	PRINT LAST NAME
ITR SIGNATURE	DATE:	PRINT LAST NAME



TS3.ID2  
ATTACHMENT 8.2

**TITLE: LICENSING BASIS IMPACT EVALUATION**

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**Section 4: Environmental Protection Evaluation**

- |    |   | Yes                      | No                       |
|----|---|--------------------------|--------------------------|
| 1. | Will there be discharges to water resulting from this activity or CTE which are either not permitted or could impact the NPDES Permit?<br>Explanation:  | <input type="checkbox"/> | <input type="checkbox"/> |
| 2. | Will any work on the SLO-2 site violate current protection requirements?<br>Explanation:  | <input type="checkbox"/> | <input type="checkbox"/> |
| 3. | Does this activity or CTE require a change to the Environmental Protection Plan (Facility Operating License DPR-80 and DPR-82, Appendix B)?<br>Explanation:   | <input type="checkbox"/> | <input type="checkbox"/> |
| 4. | Will there be discharges to air resulting from this activity or CTE which are not permitted by applicable air quality regulations and existing air pollution control permits or could impact these permits?<br>Explanation: | <input type="checkbox"/> | <input type="checkbox"/> |
| 5. | Will hazardous materials be used or stored or their quantities changed as a result of this activity or CTE?   | <input type="checkbox"/> | <input type="checkbox"/> |
| 6. | Will there be hazardous waste streams generated as a result of this activity or CTE?<br>Explanation:  | <input type="checkbox"/> | <input type="checkbox"/> |



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**TITLE: LICENSING BASIS IMPACT EVALUATION**

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Based on the above evaluation, the preparer and ITR have determined that an unreviewed environmental question

is\*     is not involved. A decrease in the effectiveness of, or a change to, the Environmental Plan

is\*     is not involved. Further, an environmental permit or permit revision  is\*\*     is not required.

\* If a UEQ or a decrease in the effectiveness of, or a change to, the Environmental Protection Plan is involved, NRC approval is required prior to implementing the activity or CTE.

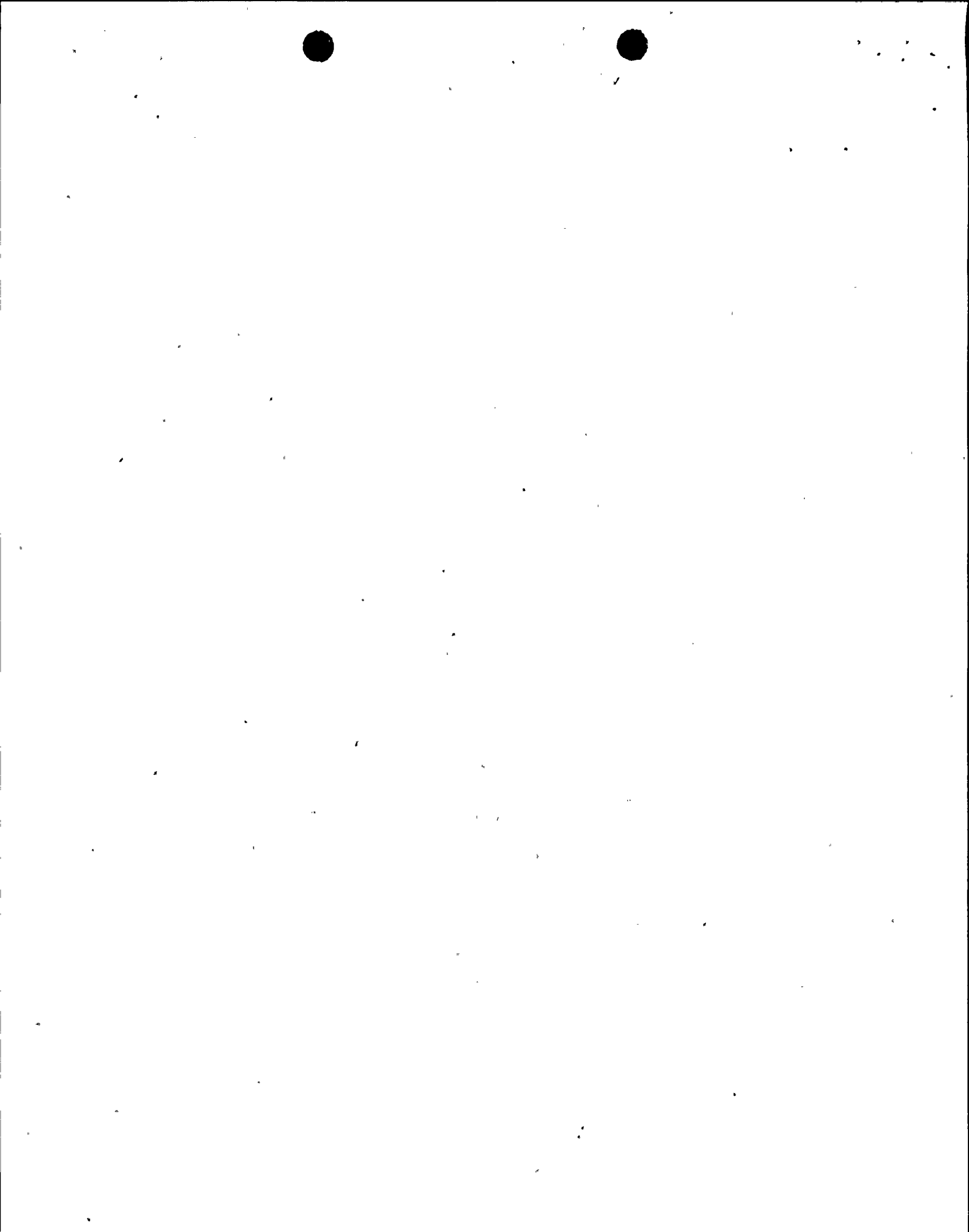
\*\* If a permit (revision) is required, list below the permit(s) or revision(s) required. Explain if the activity or CTE can commence before the permit (revision) is issued.

Environmental Protection Plan Evaluation Signatures		<input checked="" type="checkbox"/> N/A
PREPARER SIGNATURE	DATE:	PRINT LAST NAME
ITR SIGNATURE	DATE:	PRINT LAST NAME













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### 1.0 Definitions of DCP Design Seismic Events

<u>Event Magnitude (g)</u>	<u>PG&amp;E Definition</u>	<u>NRC Definition</u>
0.2	DE	OBE
0.4	DDE	NA
0.75	Hosgri	SSE

Reference: SSER 7, Section 2.5.2, 5/26/78, pages 2-3 to 2-5

### 2.0 Acceptance of RG 1.61 Damping Values for DCP

Annotated Excerpt from SSER 7, 5/26/78 (emphasis added):

#### 3.9.3.2 General Methods of Analysis Description of Methods

The methods used in the Hosgri event reevaluation of mechanical equipment and systems are given in Amendment 50 and subsequent amendments to the operating license application and are summarized below.

Floor response spectra were utilized as described in Section 3.9.3.3 below.

The methods used in the reevaluation were generally the same as used in the original analysis with the exception of the items listed below. (Listing the differences from the original methods is a way of describing the general analysis methods. Our evaluation basis is, however, the conservatism of the methods rather than comparison with the original methods.)

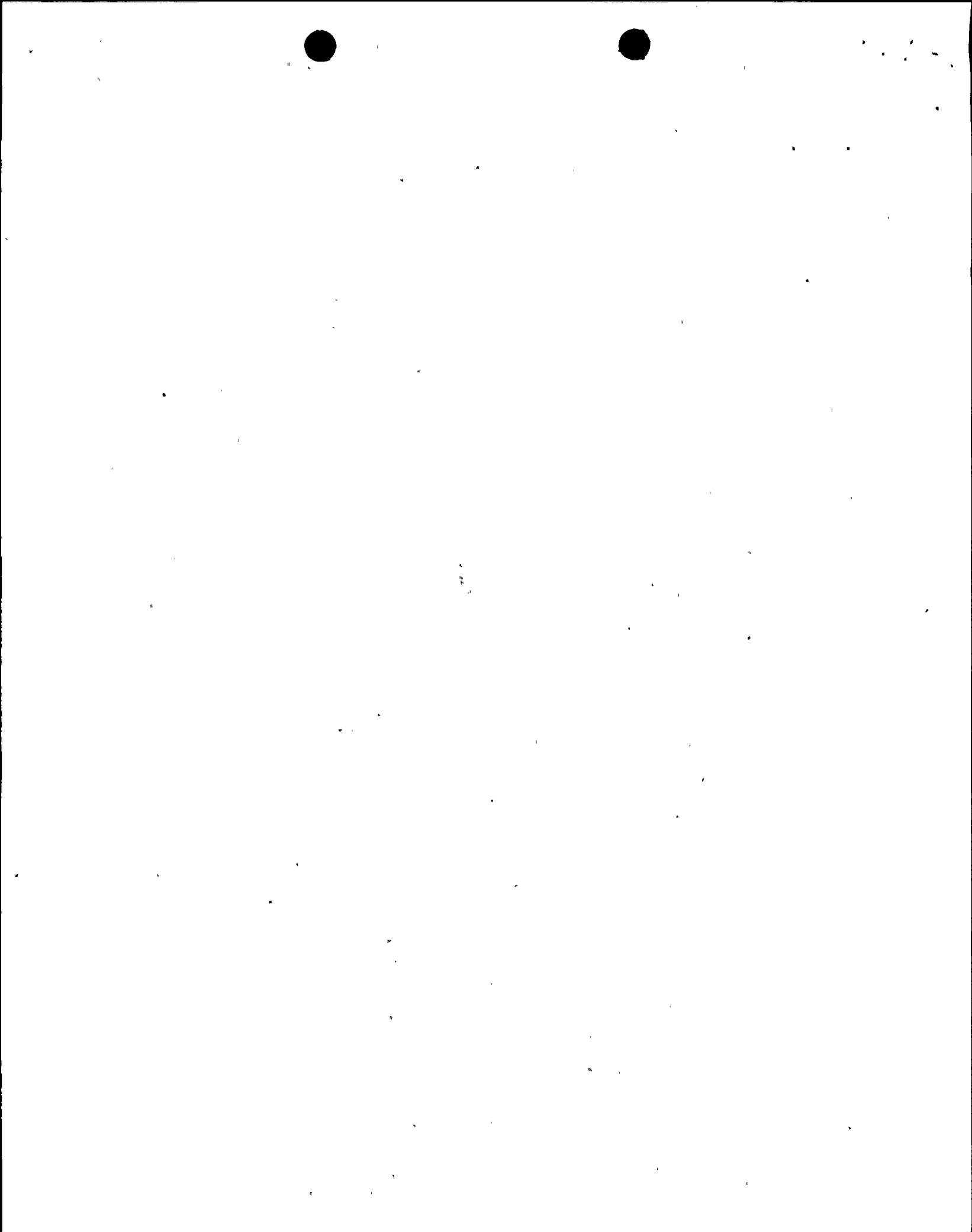
- (1) **Damping values recommended in Regulatory Guide 1.61 were generally used in the reevaluation. A damping value of 4 percent was used for the reactor coolant system as opposed to 3 percent in the Regulatory Guide.**  
*Remark: 3% is the SSE value recommended in RG 1.61 for Equipment and piping systems greater than 12" diameter. 4% is the SSE value recommended in RG 1.61 for Welded steel structures. Note that 2% damping is recommended in RG 1.61 for both these structure / component categories.*

**The damping values which were used in the original analysis for double design earthquake are below the values currently recommended in Regulatory Guide 1.61 and would give higher calculated responses.**

*Remark: "Original analysis for DDE" means the 0.4g event that was originally analyzed at 1% damping. RG 1.61 recommends 4% for DDE. Also note statement that analysis at 1% damping would produce larger response than 4%. This speaks to the concept that larger displacements justify use of larger damping values.*

**The damping values recommended by Regulatory Guide 1.61 constitute our current criteria and have been acceptable on all applications for several years. Therefore, they are acceptable for use in the seismic reevaluation.**

*Remark: This clearly states that RG 1.61 damping is approved for use in DCP analysis of mechanical equipment and systems.*



The value of 4 percent for the reactor coolant system was justified in actual plant tests by Westinghouse Electric Corporation and has been accepted in our review of other plants. The results were reported in Westinghouse Topical Report, WCAP 7921-AR, "Damping Values of Nuclear Power Plant Components", submitted in August 1973. After reviewing the topical report we approved the 4 percent value for use in similar Westinghouse reactor coolant systems provided that similarity to the system tested is demonstrated. Our evaluation was presented in a letter to Westinghouse dated May 16, 1974. We will require that the applicant demonstrate similarity and we will provide our evaluation of this item in a future supplement to the Safety Evaluation Report.

*Remark: This emphasizes validity of the WCAP-7921-AR results.*

For the reasons discussed above, the damping values used in the seismic reevaluation for mechanical equipment and systems are acceptable, subject to satisfactory demonstration of the reactor coolant system's similarity to the system that was tested.

*Remark: 4% damping is acceptable for the Hosgri event, pending verification that DCPD RCLoops are similar to those tested (i.e., Indian Point 2). The PG&E verification data is later accepted in SSER 8 of 11/78.*

Annotated Excerpt from SSER 8, 11/15/78 (emphasis added):

3.9            Mechanical Systems and Components  
3.9.3        Seismic Reevaluation  
3.9.3.1      Summary of Staff Review

In Supplement Number 7 to the Safety Evaluation Report we summarized our review of the seismic reevaluation. Since then we have continued our review in a similar manner. We have reviewed additional detailed information such as calculation sheets and test data and discussed this information with the applicant. In addition, we have reviewed the letters and amendments to the operating license application submitted by the applicant.

The following paragraphs describe our evaluation of the outstanding matters identified in Supplement Number 7, which are now resolved.

3.9.3.2      General Methods of Analysis

In Supplement Number 7 we found the general methods of analysis acceptable, subject to satisfactory demonstration of the similarity between the Diablo Canyon reactor coolant system and the system that was tested to justify 4 percent damping.

*Remark: Restatement only.*

The applicant has indicated that the Diablo Canyon loop piping is made from similar material and has approximately the same overall dimensions, dynamic characteristics, and structural stiffness as the test plant as Indian Point 2. The primary components for Diablo Canyon are manufactured from the same material, using the same fabrication methods, and have similar dynamic characteristics to the test plant primary components. The primary component supports for Diablo Canyon are similar to the test plant supports which consist of welded and bolted steel





structural elements designed to allow for free thermal expansion but restrict the motion design results in the same types of damping mechanisms. We agree that these factors indicate the Diablo Canyon reactor coolant system is similar to the reactor cool system that was tested. Furthermore, the Indian Point 2 shaker tests have demonstrated that the magnitude of damping increases with the response amplitude, independently of any frequency changes in the system. The dynamic responses of the Diablo Canyon primary cooling system and components as calculated for the Hosgri earthquake are much higher than those obtained during the tests at the test plant. This indicates that the damping value to be expected at Diablo Canyon for a Hosgri event is even greater than the value justified by the tests.

*Remark: Reaffirms concept that higher damping is appropriately applied to higher-response systems.*

Based on these factors we conclude that the use of four percent damping for the Diablo Canyon reactor coolant system analysis is acceptable.

*Remark: No remark necessary.*

We consider this matter resolved.

### 3.0 Conclusion

NRC acceptance of the RG 1.61 damping values for analysis of DCCP mechanical equipment and systems has been clearly indicated as follows:

1. Acceptance of 2% DE damping is acknowledged in SSER 7, Section 3.9.3.2 (1) third paragraph where blanket approval of RG 1.61 is explicitly stated. This approval therefore includes 4% DDE (per PG&E definition) and 4% Hosgri (DDE per NRC definition).
2. Acceptance of 4% for DDE is further acknowledged in SSER 7, Section 3.9.3.2 (1) second paragraph where the original analysis of the 0.4g DDE at 1% damping is found analyzed at below-recommended values.
3. Acceptance of 4% damping for the Hosgri is explicitly acknowledged in SSER 8, Section 3.9.3.2 third paragraph.



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