NUREG-0675
Supplement No. 9

Safety Evaluation Report

related to the operation of Diablo Canyon Nuclear Power Station Units 1 and 2

Docket Nos. 50-275 and 50-323
Pacific Gas and Electric Company
Supplement No. 9

June 1980

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

Available from

GPO Sales Program
Division of Technical Information and Document Control
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Printed copy price: $3.75

and

National Technical Information Service
Springfield, Virginia 22161

NUREG-0675
Supplement No. 9

Safety Evaluation Report

related to the operation of Diablo Canyon Nuclear Power Station Units 1 and 2

Docket Nos. 50-275 and 50-323
Pacific Gas and Electric Company
Supplement No. 9

June 1980

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

Previous supplements in this series were identified solely by Docket Number and Supplement Number. Future supplements will carry the NUREG number now assigned: NUREG-0675.

TABLE OF CONTENTS

PRE-TMI-2 ISSUES

CONTENTS (Continued)

APPENDICES

1.0 INTRODUCTION

1.1 General

The Commission's Safety Evaluation Report in the matter of Pacific Gas and Electric Company's application for operating licenses for the Diablo Canyon Nuclear Power Station, Units 1 and 2, was issued on October 16, 1974. In the Safety Evaluation Report (SER), it was stated that supplemental reports would be issued to update the Safety Evaluation Report in those areas where the staff's evaluation had not been completed. Supplement Numbers 1, 2, 3, 4, 5, 6, 7, and 8 to the SER, issued on January 31, 1975, May 9, 1975, September 18, 1975, May 1, 1976, September 10, 1976, July 14, 1977, May 26, 1978, and November 15, 1979, respectively, documented the resolution of certain outstanding items and summarized the status of the remaining outstanding items.

In SER Supplement 8, we stated that with the exception of 18 items, all safety matters identified up to that time were considered resolved. The 18 items which are summarized in Section 22.0 of SER Supplement 8 required the staff to confirm compliance with our requirements prior to the issuance of operating licenses.

In this supplement, we have provided our evaluation of non-TMI-2 accident related matters to support licensing Diablo Canyon Units 1 and 2 for fuel loading and operation up to five percent power for purposes of low-power testing. A separate safety evaluation covering TMI-2 related requirements is being concurrently prepared to support the same licensing action for Diablo Canyon Units 1 & 2. Similar licensing actions have been taken on by Sequoyah Unit 1, North Anna Unit 2, and Salem Unit 2 plants earlier this year following Commission approval.

This supplement provides our evaluation of the confirmatory actions identified in SER Supplement 8 submitted by the applicant and provides our evaluation of other safety issues that have arisen since the issuance of SER Supplement 8, not related to the new requirements based on post-Three Mile Island 2 (TMI-2) recommendations. These items, referred to as non-TMI matters, are listed in Sections 1.1.1 and 1.1.2.

The principal outstanding non-TMI-2 matter in our review of this operating license application has been the earthquake capabilities of the Hosgri fault and its impact on seismic design considerations for this plant. As a result of this matter, the applicant has performed a reevaluation of the plant's seismic capabilities as reported in previous supplements to determine what modifications are necessary in order to demonstrate that the plant can withstand a more severe earthquake than was considered in the original design. As discussed in Section 3.0 of SER Supplement 8, we concluded that the plant's capability to withstand the design basis earthquake was acceptable. The Atomic Safety and Licensing Board (ASLB) concluded hearings on seismic issues on February 15, 1979, and issued a partial initial decision, dated September 27, 1979, finding the seismic design of the Diablo Canyon nuclear

power plants adequate. This decision was contested by intervenors and was the subject of a prehearing conference before the Atomic Safety and Licensing Appeal Board (ASLAB) on April 3, 1980, and a decision is currently pending before the ASLAB. Furthermore, there will be a re-hearing on security issues before the ASLAB during the Summer of 1980. The staff has not changed its previous conclusions on the adequacy of the seismic design and physical security plan. The applicant has completed the installation of the modifications that were indicated by the seismic reevaluation.

The Advisory Committee on Reactor Safeguards (ACRS) requested the applicant to evaluate the consequences of failure of non-seismic equipment and piping interacting with safety systems following an earthquake to determine if the Diablo Canyon plants can be safely shut down following such a postulated accident. The applicant by letters dated May 7 and May 27, 1980, submitted their response to this matter and it is currently being evaluated by the NRC staff. A separate safety evaluation report covering this matter is scheduled to be issued in August 1980.

1.1 Summary of the Non-TMI Matters

The listings in Sections 1.1.1 and 1.1.2 below identify whether the issue has been acceptably resolved for full power operation or whether the issue has been acceptably resolved only for the low power testing program. The sections of this supplement where these items are discussed is noted parenthetically for each item.

1.1.1 Confirmation of Compliance with Approved Requirements for Those Matters Identified in Supplement 8

	Approved For Full Power Operation	Approved For Low Power Test Program Only
Minor piping modification (Section 3.2.1)	X	
Containment base slab analysis (Section 3.8.5.4.1)	X	
Intake structure stability analysis (Section 3.8.5.4.4)	X	
Polar gantry crane inside containment (Section 3.8.5.4.8)	X	
Some pipe supports to withstand loads due to containment tilt (Section 3.9.3.4.4)	X	
Seismic qualification of valves (Section 3.9.3.7)	X	
Several aspects of seismic qualification of electrical equipment (Section 3.10)	X	
Capability for cold shutdown despite assumed failure of nonseismic fire protection equipment (Section 9.6.1)	X	
Several matters related to environmental qualification of Class 1E electrical equipment (Section 7.8)	X	

	Approved For Full Power Operation	Approved For Low Power Test Program Only
The means of protecting the reactor coolant system from overpressurization transients at low temperature for Unit 1 in the long term (after the first fuel cycle) and for Unit 2. (Section 5.2.2 of Supplement No. 6).	X (For Unit 1 only)	
Analysis of a postulated steam line break inside containment (Section 6.2.1)	X	
Several matters related to the plant's fire protection capabilities (Section 9.6.1)	X	
Normal containment purge system in light of our current criteria (Section 6.2.3)		X
Reanalysis based on a calculational error in the emergency core cooling calculations related to metal-water reaction heat release (Section 6.3)	X	
An evaluation of additional detailed information about reactor vessel fracture toughness properties (Section 5.2.4)	X	
Confirmation of compliance with our requirements for documentation concerning RHR valve position indication (Section 7.6)	X	
An evaluation of the vulnerability of the electric power systems and equipment to a degraded voltage condition (Section 8.0)	X	
Fault current protection for containment electrical penetrations (Section 8.1)	X	

1.1.2 New Safety Issues That Have Arisen Since Issuance of Supplement Number 8:

1.	Aging Considerations for Batteries (Section 3.10.6)	X	
2.	Refueling Water Storage Tank Capacity (Section)	X	
3.	Thermal Performance Analysis (Section 4.2)		X
4.	Internal Fuel Rod Pressure (Section 4.2)	X	
5.	Grid Straps (Section 4.2)	X	
6.	Rod Cluster Control Assembly Spiders (Section 4.2)	X	
7.	Guide Tube Wear (Section 4.2)	X	
8.	Rod Drop Transient (Section 4.2)	X	
9.	Secondary Water Chemistry (Section 5.7.2.1)	X	
10.	Steam Generator Inspection Ports (Section 5.7.2.2)	X	
11.	Row 1 Steam Generator Tubes (Section 5.7.2.3)	X	
12.	Habitability Systems (Section 6.4)	X	
13.	Steam Generator Level Instrumentation (Section 7.8)	X	
14.	Environmental Qualification Documentation (Section 7.8)		X items identified paragraph h(1) and concluding paragraph in Section 7.8

	Approved For Full Power Operation	Approved For Low Power Test Program Only
15. Acceptability of Operating Procedures to Mitigate the Consequences of an Anticipated Transient Without Scram (Section 15.0)		X
16. Quality Assurance (Section 17.0)	X	

Exemptions are required from Appendices G, H, and J of 10 CFR Part 50. Our safety evaluations supporting these exemptions are discussed in Sections 5.2.4 and 6.2.6 of this report respectively.

Appendix A to this SER supplement is a continuation of the chronology of the principal events involved in the Commission staff's radiological safety review.

Appendix B to this SER supplement provides additional information on generic safety issues to supplement the information provided to the ASLB on Diablo Canyon on February 5, 1979.

3.0 DESIGN CRITERIA-STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.2 Classification of Structures, Components, and Systems

3.2.1 Seismic Classification

In SER Supplement 8, we stated that we would require that the piping between the backup auxiliary feedwater supply from the fire water storage tank and the auxiliary feedwater system be designed to withstand the Hosgri event (seismic Category I) and have parallel isolation valves to ensure operability in the event of a single active failure.

In Amendment 78 to the Final Safety Analysis Report, the applicant confirmed that the piping between the fire tank and the auxiliary feedwater system was designed to seismic Category I criteria and has parallel valves to ensure operability in the event of a single active failure. Based on the seismic Category I design and the redundant valving in the line, we find this piping system now acceptable.

3.8.5.4 Structural Analysis

Since the time of issuance of the SER Supplement 8; we have reviewed additional information submitted by the applicant regarding the confirmation of structural acceptability of the following items:

3.8.5.4.1 Containment Base Mat

In SER Supplement 8, we stated that we have reviewed the preliminary submittal of the method of the analysis of the containment base mat, including the analytical model, and we found the approach conservative and thus acceptable. Since then, the applicant submitted his final calculations, and we found that all positive and negative moments are within the capacity of the foundation mat. Based on this confirmatory information, we consider this matter resolved.

3.8.5.4.4 Stability Analysis of Intake Structure

In SER Supplement 8, we stated that at the meeting on November 8, 1978, the applicant presented an analysis of stability of the intake structure and we and our consultants agreed with the applicant's approach. Since that time, the applicant submitted the results of this analysis which indicate that there is a factor of safety of 2.65 against overturning. The applicant also investigated the foundation soil-bearing pressure under the intake structure and reported that its maximum value is 6.92 ksf which gives a factor of 4.7 when compared with the allowable soil-bearing pressure. Based on this confirmatory information, we consider this matter resolved.

3.8.5.4.8 Cranes

In SER Supplement 7, we stated that there are six cranes in the plant which could have adverse effects on safety-related equipment in case of a failure due to an earthquake.

Since that time, we reviewed the information regarding the applicant's structural analysis which was performed using the guidelines developed by the Regulatory staff. The description of the methods of the analyses and our evaluation of the results for individual cranes is provided in the following paragraphs. As noted below, we conclude that there is reasonable assurance that these cranes will perform their required safety functions in the case of a postulated seismic event.

Fuel Handling Building Crane

The fuel handling building crane is located in the fuel handling area of the auxiliary building. Its capacity is 125 tons. The analysis was performed for the postulated Hosgri event with the crane in both the loaded and unloaded condition and in parked and various operating positions. The criteria for stress evaluation were the same as for other Category I structures.

Linear three-dimensional finite element analyses were performed using an appropriate response spectrum as the input. The analyses were performed using the SAP IV structural analysis computer code. Seismic loading in the horizontal directions were the elastic floor response spectra for 7 percent of critical damping for the auxiliary building, which incorporated the torsional effects. These spectra were developed from the Blume and Newmark horizontal response spectra. Seismic loading in the vertical direction was the elastic floor response spectrum for 7 percent critical damping developed from two-thirds of the Newmark 0.75g horizontal free-field response spectrum. Since the crane and the building are primarily bolted structures, we found that the use of 7 percent damping is appropriate.

Due to the orientation of the crane (the crane girder is in the north-south direction), the east-west response has no significant effect on the crane itself. In the north-south dynamic analyses, a single-bent frame model was developed using the SAP IV computer code, considering the total mass of the trolley lumped on one girder. In the case with operating load a non-linear, vertical seismic analysis of the crane was performed using the DRAIN-2D computer code. The input motion was applied at elevation 170 feet in form of time history as derived from the SAP IV dynamic analysis and the use of the MATRAN computer code. The analysis was made for the rated operating load of 125 tons on the main hoist and 15 tons on the auxiliary hoist, with structural damping of 4 percent critical in the first two modes of the analysis. The analysis took into account the pendulum motion of the suspended load using the DRAIN-2D computer code and the response spectrum method.

The square-root-of-the-sum-of-the-squares (SRSS) method was used to combine the seismic responses in each mode and the three earthquake directions (E/W, N/S and vertical). Maximum

bending moments and axial loads resulting from the SRSS combined effects of the separate response spectrum analyses were added directly to the dead load effects for the unloaded and loaded cases.

Review of the results of structural analyses show that, with the structural modifications that have been made (namely, installation of clamps on the trolley to prevent dislocation from the tracks and providing a bumper to prevent the load to be dislocated longitudinally), the crane building system is adequate during the postulated Hosgri seismic event for three cases: the unloaded case, with the rated operating load of 15 tons on the auxiliary hook, and 125 tons on the main hook.

Sliding of the crane of a few inches can be expected along the trolley and crane runways during the postulated Hosgri earthquake. However, the effect is inconsequential because the stresses are within the allowable stresses.

Since the elevation of the center of gravity of the crane coincides with the top of the crane rails, overturning of the crane is impossible.

Predicted seismically induced relative displacements for the unloaded and loaded crane, relative to the crane supports, are found to be approximately 1.0 in. and 1.5 in. in the north-south and vertical directions, respectively.

For any member, the earthquake effects are combined on an SRSS basis and added directly to the dead load effects to obtain the combined stress effect. The maximum combination of the ratios of the computed bending stress to allowable stresses results in a ratio of 0.30 for the unloaded case and 1.00 for the 125-ton loaded case.

Polar Crane

There are two polar cranes at the Diablo Canyon Plant, one located in each containment. These are gantry type cranes with trolleys.

The polar cranes are used for reactor head and equipment movement within the containment.

We reviewed the information submitted by the applicant, described below, and concluded that for both the seismically locked and unlocked condition, the polar cranes remain stable with stresses within the prescribed acceptance criteria during the postulated Hosgri event. Thus, the polar cranes can be safely operated without restriction or modification.

In the unlocked condition, which allows the crane to be in any position within the containment, two crane hold-down conditions were considered: (1) the "free" condition (an upper bound condition) which did not include the seismic hold-down clamps; and (2) "tied" condition which included the seismic hold-down clamps.

For both cases, the model was free to rock during the nonlinear dynamic response. For the case with operating load, the cable was modeled as a nonlinear truss element with zero buckling strength in order to simulate impact effects of the cable-suspended load.

The seismic input consisted of the acceleration histories developed for the 140-ft elevation of the containment structure, which is also top of the crane railing, from the Newmark response spectra.

The transverse model, incorporating the nonlinear gap and truss elements, was subjected to simultaneous horizontal and vertical seismic motions to obtain peak axial loads and transverse bending moments in the crane legs. The longitudinal model was subjected to the same motions to obtain peak bending moments in the crane girders and longitudinal moments in the crane legs. The separate seismic effects were combined on an SRSS basis and added directly to gravity effects. Resulting stresses were compared with allowable values.

Results show that the polar cranes are structurally adequate in the parked and seismically locked condition during the postulated Hosgri seismic event. The cranes remain stable with stresses within the prescribed acceptance criteria.

The structural criteria used are the same as those described for the fuel handling building crane.

For parked and locked condition methods of analysis used were similar to those described for the fuel handling building crane. The mathematical models, response spectra and other specific details used are those applicable to the polar cranes. These spectra correspond to the Newmark elastic floor response spectra for 4 percent damping at elevation 140 feet in the containment. Since the cranes are bolted structures for which Regulatory Guide 1.61 allows 7 percent damping, appropriate scaling factors were applied to stress ratios based on 4 percent damping. We reviewed the method by which the scaling factors were computed, and we found it acceptable.

Nonlinear analyses were performed to determine the effects of the response on the stability of the crane and the stresses in the individual structural members.

The polar cranes were modeled as two-dimensional nonlinear frame structures using the DRAIN-2D computer code. The structures consisted principally of beam-column elements. The support points were modeled as nonlinear gap elements. Maximum bending moments and axial loads for the crane legs, girders, end ties, and cross beams that result from the SRSS combined effects of the separate response spectrum analyses were added directly to the dead load effects. For any member, the ratios of the computed bending moment and axial stresses to allowable values are additive to obtain the combined stress effect. None of the combinations resulted in a ratio greater than 1.0.

The maximum support reactions resulting from the SRSS combined effects of the separate response spectrum analyses were added directly to the dead load effects.

For unlocked condition, two-dimensional nonlinear analyses of both the free and tied conditions were performed. In each case, the cranes were considered as both loaded and unloaded.

For the free unloaded crane, maximum uplift was approximately 3 in. for the transverse excitation and 2-1/2 in. for the longitudinal excitation. Maximum relative horizontal displacements at the top of the crane (elevation 205 ft.) were approximately 10 in. in both the transverse and longitudinal direction.

For the free 200-ton loaded condition, maximum uplift was approximately 4-1/2 in. in transverse rocking. There was no uplift associated with longitudinal response. Maximum horizontal displacements at the top of the crane were approximately 13 in. and 10 in. in the transverse and longitudinal directions, respectively. The maximum lateral displacement of the crane girder was calculated to be approximately 13 in. Pendulum motion of the load would result in a peak displacement of nearly 18 in.

Maximum bending moments and axial loads for the crane legs, girders, end ties, and cross beams that result from the SRSS combined effects of the separate components of seismic excitations added directly to the dead load effects for the unloaded and loaded condition. For any member, the ratios of the computed bending moment and axial stresses to allowable values, the total interaction ratio is the sum of that due to dead load and the SRSS combination of those due to the separate earthquake effects. None of the combinations result in a ratio greater than 1.0.

The preliminary and more detailed response history nonlinear analyses based on energy considerations indicated that the crane would remain stable during the postulated Hosgri event.

For the tied unloaded crane, maximum uplift was approximately 1-1/2 in. due to transverse and longitudinal rocking, respectively. Maximum horizontal displacements at the top of the crane (elevation 205 ft.) were approximately 6 in. and 9 in. in the transverse and longitudinal directions, respectively.

For the tied, 200-ton loaded condition, maximum uplift was approximately 1 in. in transverse rocking. There was no uplift associated with longitudinal response. Maximum horizontal displacements at the top of the crane were approximately 7-1/2 in. and 10 in. in the transverse and longitudinal directions, respectively.

Maximum bending moments and axial loads for the crane legs, girders, end ties, and cross beams that result from the SRSS combined effects of the separate components of seismic excitations were added directly to the dead load effects for the unloaded and loaded conditions. For any member, the ratios of the computed bending moment and axial stresses to allowable values, the total interaction ratio, is the sum of that due to dead load and the SRSS combination of those due to the separate earthquake effects. None of the combinations result in a ratio greater than 1.0. The factor of safety against overturning is found to be almost six.

On the basis of our review of the methods and of the results of the seismic evaluation of the polar cranes, we concluded that the cranes, without restriction or modification, can be operated without undue risk to the health and safety of the public.

Turbine Building Cranes

There are two bridge-type cranes located in the turbine building. These cranes have trolleys which travel in a direction perpendicular to the travel of the bridge. The cranes span the width of the turbine building and are approximately 40 feet above the operating deck of the turbine building.

The turbine building cranes are used for moving equipment during inspection and maintenance and may be used during plant operation. Analyses show that the cranes and turbine building are structurally adequate for loads of 100 tons or less during the postulated Hosgri event, with the restriction that both cranes cannot be located in the same unit.

The only safety-related items located in the turbine building are the diesel generators, 4kV switch gear, cardox tank, and component cooling water heat exchangers.

Results of preliminary structural analyses showed that the cranes were adequate for loads up to 100 tons, but that the turbine building exterior columns were limited to crane loads of 15 tons or less.

Additional structural analyses, discussed in the next subsection, identified the modifications necessary to qualify the turbine building for crane loads up to 100 tons. These modifications have been completed and are identified in this report.

The structural criteria used are the same as those described for the fuel handling building crane and other Category I structures.

Methods of analysis used were similar to those described for the fuel handling building crane. Detailed mathematical models, response spectra, and other specific details used are those applicable to the turbine building cranes. The cranes were modeled as part of the building. Finite elements models for the linear and nonlinear analyses were generated. Horizontal spectra for the crane/building model were the Blume and Newmark horizontal elastic spectra for 7 percent damping. The time histories corresponding to these spectra were increased by 10 percent to account for torsion. The vertical input spectrum for the crane/building model was two-thirds the Newmark 0.75g horizontal free-field response spectrum.

North-south dynamic analyses of the planar turbine building wall, both with and without the crane mass, were performed using the TABS computer code to obtain mode shapes, frequencies, and participation factors. This provided the input for a MATRAN analysis (mathematical manipulation) whereby a north-south response spectrum and time history were generated at elevation 180 ft., the crane rail. The crane model was then subjected to a response spectrum analysis using the derived response spectra at elevation 180 ft., and structural damping of 4 percent for the unloaded crane to obtain north-south response. In accordance with Regulatory Guide 1.61, damping of 4 percent was used because the crane is primarily a welded structure.

In this model, the total trolley mass was lumped on one girder for lateral response. Lateral framing action provided by the two crane griders and end ties was also considered.

Linear dynamic analyses of the crane/building model were performed using the SAP IV structural analysis computer code. The results were used in conjunction with the SPECTH computer code to obtain the acceleration time histories of the bridge girder support nodes at elevation 180 ft.

Nonlinear vertical analyses of the loaded crane were then performed using this motion as input to the DRAIN-2D computer code. Structural damping of 4 percent in the first two modes was adopted for these analyses.

The horizontal pendulum motion of the suspended load was determined using both the DRAIN-2D nonlinear time history analysis method and the SAP/SPECTH response spectrum method.

Results of the linear and nonlinear analyses show that, except for a few structural modifications, the crane/building systems is adequate during the postulated Hosgri seismic event for the unloaded case and for an operational load up to 100 tons.

Some modifications of the crane were necessary to transmit the transverse and upward forces directly from the trolley to the crane bridge girders and from the crane bridge to the crane runway girders. These modifications have been completed.

Analysis also showed that structural modifications were necessary to qualify the turbine building exterior columns to allow the crane to carry a 100-ton load during the postulated Hosgri event. The modifications, which have been completed, consist of strengthening 26 of the 54 exterior columns by welding additional material to the columns between elevations 130 feet and 150 feet. These building modifications, together with the above mentioned minor crane modifications, allow the use of one turbine building crane in each unit for loads up to 100 tons. These modifications have been completed in Units 1 and 2.

The maximum predicted seismic-induced displacements, relative to the crane supports, are approximately 2 in. and 5 in. in the north-south and vertical directions, respectively.

The ratio of the computed bending moment and axial stresses to allowable values for the main crane members with a 100-ton load due to earthquake effects were combiined for each member, on an SRSS basis and added directly to the dead load effects to obtain the combined stresses. None of the combinations had a ratio greater than 1.0.

Because the center of gravity for the turbine building cranes coincides with the top of the crane rails, the overturning of the crane is impossible.

On the basis of the above, we concluded that the results of the seismic evaluation of the turbine building cranes demonstrate that the cranes comply with the NRC staff's guidelines and that the cranes can be operated without undue risk to the health and safety of the public. Furthermore, we concluded that the structural analysis shows that for the unloaded

and loaded (100 tons or less) conditions, the cranes remain stable and stresses in the cranes and turbine building remain within the prescribed acceptance criteria during the postulated Hosgri event.

Intake Structure Crane

The one intake structure crane at the Diablo Canyon plant is of the gantry type with a trolley which travels in a direction perpendicular to the direction of travel of the gantry.

The only safety-related items that could potentially be affected by an intake structure crane failure would be the auxiliary salt water pumps. In case of a seismically induced failure of the intake structure crane resulting in a damage of the auxiliary saltwater pumps, only those pumps for one unit could be affected. If crane failure is assumed to disable auxiliary saltwater pumps for one unit, the auxiliary saltwater pumps from the other unit can also provide service to the damaged unit via a cross tie line. In addition to this redundancy between the two auxiliary saltwater systems, long-term core cooling can be provided using the steam generators and the auxiliary feedwater system.

The structural criteria used for the intake structure crane are the same as those described for the fuel handling building crane.

Methods of analysis used are similar to those described for the fuel handling building crane. Detailed mathematical models, response spectra, and other specific details are those applicable to the intake structure crane.

Seismic motions were input into the crane structure at the crane rail. Seismic loading in the horizontal directions were the Newmark and Blume elastic spectra for 7 percent damping in the east-west and north-south directions respectively. Seismic loading in the vertical direction was two-thirds the Newmark free-field elastic spectrum for 7 percent damping. We agreed with the applicant that this value is appropriate because this crane is primarily a bolted structure.

Results show that, except for minor modification, the crane is adequate during the postulated Hosgri seismic event for both the loaded and unloaded cases. The crane is capable of carrying its maximum rated load of 50 tons without overstressing of structural members or the hoist cable.

The maximum predicted seismic-induced displacements, relative to the base, are approximately 3 in. in the east-west and north-south directions.

Maximum bending moments and axial loads for the crane legs, girders, and end ties resulting from the SRSS combined effects of the separate response spectrum analyses are added directly to the dead load effects. For any member, the ratios of the computed bending moment and

axial stresses to allowable values are added to obtain the combined stress effec.. None of the combinations result in a ratio greater than 1.0. For the crane, the horizontal input motion was increased by 10 percent to account for torsion. Shear stresses were insignificant in all members. In the operating condition, the safety factor against overturning was found to be 8.7.

When the crane is parked at the end of the runway, stability analyses indicated that seismic overturning moments in the north-south direction may cause overstressing of the anchors. However, in this position overturning of the crane would result in the crane falling off the north end of the intake structure; this does not pose a risk to the function of any safety-related equipment.

Conclusions on Structural Analyses

As described above, we have reviewed the methods of analysis for the Containment Base Mat, Stability Analysis of Intake Structure and the structural aspects of the following cranes: the Fuel Handling Building Crane, the Polar Crane, the Turbine Building Crane and the Intake Structure Crane. On the basis of the information presented to us by the applicant, we concluded that the methods of these analyses are acceptable and that they provide reasonable assurance that in case of an Hosgri event, these structural components will perform their required safety functions.

3.9 Mechanical Systems and Components

3.9.3 Seismic Reevaluation

3.9.3.4.4 Other Piping

In SER Supplement 8, we noted one outstanding issue in our review of the piping analyses concerning the effect of containment tilting on safety-related systems connected to containment.

In SER Supplement 8, we stated that the supports for the piping connected to the containment required further evaluation to assure their integrity under the loading that would occur when the piping follows the tilting motion. The applicant has evaluated the piping supports on these lines for the Hosgri event building movement loads combined with the Hosgri event inertia loads. We have reviewed the applicant's analyses and conclude that the affected piping supports will safely withstand the calculated loads. Accordingly, we find the applicant's piping system analysis methods acceptable and consider this matter resolved.

3.9.3.7 Seismic Qualification of Mechanical Components

In SER Supplement 8, we stated that we would require the applicant to provide the documentation concerning seismic qualification of valves 9351A, PCV 455C, 8146, 8147, and HCV 142 for us to complete our review.

The valve 9351A is an air-operated globe valve. The required acceleration as installed in the plant is 0.97 g. The valve was seismically qualified by analysis for an 8.5-g acceleration, with the maximum stress 67 percent of the allowable stress. The remaining valves in question are 3-inch globe valves with air-operated actuators. In order to demonstrate the functional operability of these valves under the simulated seismic excitation, the applicant has submitted test data for similar valves 4 and 6 inches in size. These valves were tested with the same actuator size used on the 3-inch valves in the plant. We find that the similarity of the 4-inch valves tested to those used in the plant is sufficient to demonstrate the seismic qualification of these 3-inch valves. Biaxial tests of the 4-inch valves were performed at their resonant frequencies with acceleration of 4.5 g. The required acceleration for these valves as installed in the plant is less than 4. The valves were stroked open and closed during and after the test and functioned properly. We have reviewed the test data and find the seismic qualification of these valves acceptable.

In SER Supplement 8, we stated that we had not reviewed the seismic qualification of Barton Models 763 and 764 (Westinghouse Lot 1) and the Rosemount Model 1152 pressure and differential pressure transmitters. These transmitters are located below Elevation 127.00' in the containment structure. We have since completed our reviews of the information provided by the applicant concerning the seismic qualification of these transmitters. Prototypes of each of these transmitters were qualified by acceptable multi-frequency and multi-axis testing. The test input levels exceed the input levels required for the Diablo Canyon Nuclear Power Plant. The transmitters were monitored during the seismic testing and were found to function properly. We find the information provided by the applicant constitutes an acceptable resolution of this matter.

3.10.6 Summary of Outstanding Items

As described in Sections 3.10.3 through 3.10.5 of SER Supplement 8, we required that certain actions, summarized below, be taken in order to resolve matters related to seismic qualification of electrical equipment.

(1) Satisfactory completion of actions related to balance-of-plant equipment that is being retested (Section 3.10.3).

(2) Submittal of detailed qualification information for devices on instrument power a-c panelboards which are balance-of-plant equipment not to be retested (Section 3.10.3).

(3) Evaluation of recently received information concerning Barton and Rosemount pressure and differential pressure transmitters within the nuclear steam supply system scope (Section 3.10.4).

(4) Submittal of additional information concerning (a) identification and qualification of pressure and differential pressure transmitters, (b) level oscillations

in pressure and differential pressure transmitters, and (c) qualification of wide range reactor coolant temperature detectors within the nuclear steam supply system scope (Section 3.10.4).

(5) Submittal of amendments to the Final Safety Analysis Report to incorporate draft information and to eliminate inconsistencies. Also, submittal of a list of all electrical equipment requiring seismic qualification (Section 3.10.5).

These action items have now been resolved as follows:

In regard to item (1), we had identified 17 categories of equipment for which we did not find the monitoring of electrical functions adequate and defined acceptable resolutions by stating specific requirements for each category equipment. We have completed our evaluation of additional information provided by the applicant and have concluded that the applicant has complied with our specific requirements with the exception of (a) the Safeguard Relay Board and (b) the Fan Cooler Motor Controller. These exceptions have been resolved as follows:

(a) Detailed information on retesting of the Safeguards Relay Board, which had not been available for our review when SER Supplement 8 was issued, has been now incorporated into the Wyle test report. Based on our evaluation of this information, we conclude that the applicant has complied with our specific requirements for the Agastat series 7000 type relays, and therefore we find the seismic qualification of the Safeguards Relay Board to be acceptable.

(b) Chatter of the Fan Cooler Motor Controller contactor was observed during the seismic qualification testing program. To preclude contact chatter, the applicant has committed to install mechanical interlocks before licensing. We conclude that this modification is acceptable and that the Fan Cooler Motor Controller when thus modified is acceptably qualified.

As required in item (2) above, the applicant submitted of detailed qualification information regarding devices on the instrument power AC panelboards to confirm the acceptability of their seismic design. Based on our review and evaluation of this information, we now conclude that the devices mounted on the instrument power AC panelboards are seismically qualified.

In regard to item (3) above, we stated that neither model numbers of tested specimens nor references to specific qualification test reports had been provided for Barton, Fisher & Porter, and Rosemount pressure and differential pressure transmitters. We reviewed the applicant's responses in FSAR Amendment 77 that identified the instruments by model number, provided references to seismic qualification reports and summarized the test descriptions and results. We find the information provided is adequate to confirm the qualification for these instruments and is acceptable.

In regard to item (4) above, we stated that an evaluation of signal level oscillations that occurred in pressure and differential pressure transmitters during seismic testing had not been provided. We reviewed additional information from the applicant that described these oscillations and their effect on the instrumentation systems. These oscillations were attributed to mechanical motion of the force balance mechanisms in the transducers during the test. The information provided showed that the oscillations were repeatable, that the transmitters did not suffer deformation or decalibration, and that the output signal returned to its normal value after each test. Allowances in the setpoint tolerances and system design are included to accommodate the effects of these signal oscillations on the system response. We find this evaluation an acceptable resolution of this matter.

Also in regard to item (4) above, we stated that the wide range reactor coolant temperature detector installed in the plant was a model that we had not previously reviewed and requested that the applicant provide the qualification information.

We have since received and reviewed information provided by the applicant which indicates that acceptable multi-frequency and multi-axis tests were performed on prototypes of the Sostman Resistance Temperature Detectors (Sostman RTD's 11901B's) installed in the plant. The test input levels were far in excess of the input levels required for the Diablo Canyon Nuclear Power Plant. The results of the testing show that these RTD's remained operational during and after this simulated seismic testing. We find the information provided by the applicant constitutes an acceptable resolution of this matter.

In response to item (5) above, the applicant submitted Amendment 70 to the Final Safety Analysis Report (FSAR) to incorporate draft information on which we based our review. We have compared this submittal with the earlier draft information and found that this submittal is consistent with the draft information on which we based our review. Therefore, we find that this matter has been satisfactorily resolved.

Subsequent to issuance of SER Supplement 8, during the Diablo Canyon safety hearings (January 1979), the staff testified that the full implementation of IEEE-323-1974 (aging) required further development of other ancillary standards to provide guidance on specific safety-related equipment and components. With regard to batteries, an ancillary standard has since been developed (IEEE Standard 535-1979, IEEE Standard for Qualification of Class IE Lead Storage Batteries for Nuclear Power Generating Stations). During the development of this standard, it was generally accepted by industry representatives that chemical and physical deterioration due to between 10 and 20 years of aging could result in failure of the batteries during a seismic event.

Current requirements at Diablo Canyon are that batteries be replaced when their capacity drops below 80 percent of their rated capacity (approximately 20 years). This requirement is consistent with the recommendations of IEEE Standard 450-1975 and was found acceptable.

However, with the development of IEEE Standard 535-1979 and the possibility of battery failure after 10 years due to a seismic event, the applicant has been informed that we

require either replacement of the batteries after 10 years from purchase date with batteries qualified in accordance with IEEE Standard 535-1979, or performance of a qualification test that includes aging to demonstrate that the existing batteries will meet or exceed their design specifications throughout their installed life.

The staff has concluded that this requirement will be imposed as a condition to the Diablo Canyon operating license. With the imposition of this licensing condition, we consider this item resolved.

4.0 REACTOR

4.2 Fuel Mechanical Design

The fuel for Diablo Canyon, Units 1 and 2 is the Westinghouse 17x17 design. This fuel design is currently being used in fuel assemblies installed in six operating plants. Three such plants have completed the first cycle of operation, and their fuel has been inspected.

Since the time of the original Diablo Canyon 1 and 2 SER, Westinghouse has substantially changed its methods of fuel thermal performance analysis and has adopted new internal fuel rod pressure criteria. Also, at one of the operating 17x17 plants, an unexpected number of failures in two assembly components (grid straps and control spiders) was observed during refueling. These analytical changes and component failures and their impact on Diablo Canyon 1 and 2 are new issues that have arisen since issuance of SER Supplement 8 and are discussed and evaluated below.

Thermal Performance Analysis

The new Westinghouse fuel thermal performance code (PAD 3.3) is described in WCAP-8720, "Improved Analytical Methods Used in Westinghouse Fuel Rod Design Calculations," October 1976. This code contains a revision of an earlier fission gas release model and revised models for helium solubility, fuel swelling, and fuel densification.

The new Westinghouse code was approved with four restrictions as described in our safety evaluation of February 9, 1979 (Letter from J. Stolz, NRC to T. Anderson, Westinghouse). Three of those restrictions deal with numerical limits and have been complied with. The fourth restriction relates to use of the PAD-3.3 code for the analysis of fission gas release from uranium dioxide (UO_2) for power increasing conditions during normal operation. This restriction applies to the safety analysis of Diablo Canyon Units 1 & 2. However, Westinghouse has stated that this restriction does not adversely affect the results of the safety analyses performed for Diablo Canyon Units 1 & 2. Although we believe that this is essentially correct for the planned operation of Diablo Canyon Units 1 & 2, Westinghouse has prepared and submitted a detailed evaluation of this restriction. In our previous evaluation, we agreed that the PAD-3.3 code may be used for the analysis of constant high power level conditions which conservatively bound power increasing conditions during normal operation.

For operation at five percent of full power the restriction for PAD-3.3 is not significant and the analysis as presently docketed is acceptable. We will complete our review of the Westinghouse evaluation (and the applications of the revised model) prior to authorizing operation at full power.

Internal Fuel Rod Pressure

Diablo Canyon 1 and 2 now use the revised internal fuel rod pressure criteria as described in WCAP-8963A, "Safety Analysis For the Revised Fuel Rod Internal Pressure Design Basis" dated January 1979. The NRC evaluation and approval of these new criteria are also included in WCAP-8963A. The applicant has performed calculations for Diablo Canyon 1 and 2 with the approved Westinghouse fuel performance code (PAD 3.3, see above) and has shown that the internal pressure criteria are met. Therefore, we reconfirm our previous conclusions that the internal fuel pressure analysis for Diablo Canyon, Units 1 & 2 is acceptable.

Grid Straps

During a recent refueling at a similar Westinghouse 17x17 plant (Salem 1), strap damage on a number of spacer grids was observed on discharged assemblies. Similar damage has been reported previously (WCAP-8183, Rev. 1 through 8) but never to the extent observed at Salem 1, where 31 fuel assemblies suffered some damage. The damage ranged from deformed edges and small chips to loss of full strap width pieces and was usually confined to 1 or 2 of the eight grids per assembly. An evaluation for Salem, Unit 1 showed that such grid-strap damage was not detrimental to the operation of the reactor (see Amendment No. 20, October 1979, to the Salem, Unit 1 operating license, Docket No. 50-272). This evaluation considered thermal- hydraulics, neutronics, grid-cell deformation, flow blockage from loose pieces, and control-rod interference; the effects of all of these were found to be insignificant. We conclude that the Salem Unit 1 Evaluation regarding grid strap damage is also applicable to Diablo Canyon Units 1 & 2 and the effects of grid strap damage would be insignificant. On Salem, we did not evaluate return to power with assemblies that had lost full strap width pieces since the licensee elected not to return these assemblies to the core.

Westinghouse has recommended certain procedural changes that are designed to minimize or eliminate damage during fuel handling. These recommendations are based on the following: (1) loading sequence as to the buildup of rows and corner postitions in the core, (2) offset into the open regions for vertical movement of assemblies, and (3) revised load cell limits on the refueling crane to increase the sensitivity in detecting spacer grid interference. The Pacific Gas & Electric Company (PG&E) has agreed to follow these recommendations at Diablo Canyon 1 and 2 (letter from P. A. Crane, Jr., PG&E, to J. F. Stolz, NRC, dated September 28, 1979). DOR Information Memorandum No. 19 issued on October 25, 1979, also requests all licensees of 17x17 plants to visually inspect their discharged fuel for grid strap damage. Should these inspections reveal significant strap damage, further changes to the fuel handling procedures will be made. On the basis that less than full width grid strap damage is not detrimental to reactor operation and that fuel handling procedural steps will be taken at Diablo Canyon to minimize all grid strap damage, we find that this matter is satisfactorily resolved for the present.

Rod Cluster Control Assembly Spiders

Another core component failure, involving control rod spiders, was also observed at Salem Unit 1. Eight alignment fingers on six spiders failed during plant operation. Thus, eight

control rodlets became detached and were inserted into the core producing an observed flux tilt. This failure was traced to a manufacturing procedure that introduced a contaminant that led to stress-corrosion cracking of the finger. This manufacturing procedure was primarily used for two lots of fingers, and the procedure has since been corrected to eliminate the problem. A complete evaluation of this problem and its safety implications is contained in Amendment 20 to the Salem Unit 1 operating license (October 1979, Docket No. 50-272).

The evaluation agrees with the Westinghouse conclusions that:

(a) Failures do not represent a structural inadequacy or generic design weakness.

(b) Failures are the result of stress corrosion cracking and were contained within the two receiving lots of outer fingers.

(c) Elimination of all rod control clusters containing fingers from the suspect lots should prevent recurrence.

The evaluation also shows that if rodlets were dropped, the safety effects for the core would depend upon the number of dropped rodlets. A few dropped rodlets (about 10) could cause a flux tilt, but the core parameters could be maintained within the Technical Specification limits. A larger number of dropped rodlets (about 50) would be needed to cancel the excess shutdown margin or significantly affect peaking factors, but such a quantity would be easily detected and appropriate actions taken. In light of the low probability of the future occurrence of dropped rodlets and the fact that the dropping of significant number of rodlets would be detected, this matter is adequately resolved. We have reviewed the Salem evaluation and have determined that it is applicable to Diablo Canyon Units 1 & 2. Therefore, we conclude that this matter is acceptably resolved for Diablo Canyon Units 1 & 2.

Guide Tube Wear

An unexpected degradation of guide thimble tube walls has been observed during examinations of irradiated fuel assemblies taken from several operating pressurizer water reactors. Subsequently it has been determined that coolant flow up through the guide thimble tubes and turbulent cross flow above the fuel assemblies have been responsible for inducing vibratory motion in normally fully withdrawn ("parked") control rods. When these vibrating rods are in contact with the inner surface of the thimble wall, a fretting wear of the thimble wall occurs. Significant wear has been found to be confined to the relatively soft Zircaloy-4 thimble tubes because the control rod claddings -- stainless steel for Westinghouse-NSSS designs -- provide a relatively hard wear surface. The extent of the observed wear is both time and NSSS-design dependent and has, in some non-Westinghouse cases, been observed to extend completely through the guide thimble tube walls, thus resulting in the formation of holes.

Guide thimble tubes function principally as the main structural members of the fuel assembly and as channels to guide and decelerate tripped control rods. Significant loss of mechanical integrity due to wear or hole formation could (1) result in the inability of the guide thimble tubes to withstand their anticipated loadings for fuel handling accidents and transients, and (2) hinder scramability.

In response to the staff's attempt to assess the susceptibility and impact of guide thimble tube wear in Westinghouse plants, Westinghouse in letters dated September 12, 1978, December 15, 1978 and June 27, 1979, and the applicant in letter dated March 25, 1980 have submitted information on their experience and understanding of the issue. This information consisted of guide thimble tube wear measurements taken on irradiated fuel assemblies from Point Beach Units 1 and 2 (two-loop plants using 14x14 fuel assemblies).

Also described was a mechanistic wear model (developed from the Point Beach data) and the impact of the model's wear predictions on the safety analyses of plant designs such as those utilizing 17x17 fuel assemblies.

Westinghouse believes that its fuel designs will experience less wear than that reported for other NSSS designs because the Westinghouse designs use thinner, more flexible, control rods that have relatively more lateral support in the guide tube assembly of the upper core structure. Such construction provides the housing and guide path for the rod cluster control assemblies above the core and thus restricts control rod vibration due to lateral exit flow. Also, Westinghouse believes that its wear model conservatively predicts guide thimble tube wear and that even with the worst anticipated wear conditions (both in the degree of wear and the location of wear) its guide thimble tubes will also be able to fulfill their design functions. It is anticipated that some fuel elements will stay in the reactor vessel for a maximum of three to four years.

We have reviewed this information and conclude that the Westinghouse analysis probably accounts for all of the major variables that control this wear process. However, because of the complexities and uncertainties in determining (a) contact forces, (b) surface-to-surface wear rates, (c) forcing functions, and (d) extrapolations of these variables to other fuel designs (such as the 17x17 design used in Diablo Canyon), we believe that it is prudent for the applicant to consider participation in a surveillance plan for the examination of guide thimble wear.

The specifics of such a surveillance program have not yet been determined, but since the wear phenomenon is a time-dependent process, the details of such an inspection program including wear prediction do not need to be specified prior to the first Diablo Canyon refueling outage. Furthermore, such inspection may not have to be conducted at Diablo Canyon. For example, the applicant could join in a cooperative owner's group and thereby submit information derived from another plant using 17x17 fuel assemblies if sufficiently similar. For acceptability, the program should be to demonstrate that wear is within predicted limits and there will be no occurrence of hole formation in rodded guide thimble tubes for the maximum stay time in the reactor vessel without inspection (up to four years).

Since the applicant agreed by letter dated March 25 to provide results from a surveillance program as described above, this issue is adequately resolved for the first cycle of operation. This issue will be resolved for later cycles of operation by the surveillance program results. If the surveillance results do not confirm the predictions of the analysis, we will require the applicant to take appropriate action to account for the increased wear.

Rod Drop Transient

We recently completed changes to the negative rate trip Technical Specification for Diablo Canyon Units 1 and 2 to provide protection against potential power overshoots (and, hence, possibly DNB) in the event of single rod drop incidents. We had taken that action as a result of a 10 CFR Part 21 deficiency notification and corrective recommendations from Westinghouse. As part of its continuing analysis of single rod drops being performed for a topical report, Westinghouse has found several new nonconservatisms which indicate that the trip setpoint changes made earlier do not necessarily provide the desired protection. This was discussed at a meeting with Westinghouse on November 19, 1979 in Bethesda, Maryland. At the meeting Westinghouse suggested an interim procedural position which would provide protection in single rod drops. The staff approved this position until a long term solution to the problem can be developed. The position is as follows:

(a) The plant may operate in manual control from 0-percent to 100-percent power with no changes in the current rod insertion limits.

(b) The plant may operate in automatic control from 0-percent to 90-percent power with no changes in the current rod insertion limits; above 90-percent power the D control rod bank would have to be withdrawn to 215 steps or greater.

In a letter dated March 17, 1980 to J. Stolz, PG&E Co., has agreed to implement these restrictions in Diablo Canyon Units 1 and 2. The basis for our finding the interim position acceptable is that it prevents an overshoot above full rated thermal power in the event of a dropped rod. For power levels equal to or greater than 90 percent in automatic control, a dropped rod event will result in a withdrawal demand from the rod control system. Since differential rod worth of the D bank while above 215 steps is negligible, the reactivity required for a power overshoot is not available. For rod drops below 90 percent power in automatic control, analysis by Westinghouse shows that the reactor will not overshoot the above rated power. In manual control, the operator will not react to cause a power overshoot. Thus, the DNB design limit is not exceeded and, consequently, we find the interim position acceptable.

5.0 REACTOR COOLANT SYSTEM

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.4 Fracture Toughness

Compliance with Appendices G and H, 10 CFR Part 50

Pacific Gas and Electric Company stated that the requirements of Appendices G and H of 10 CFR Part 50 were met for Unit Nos. 1 and 2 except for the specific requirements of Sections III.C.2 and IV.A.4 of Appendix G and the specific requirement of Section II.B of Appendix H for Unit No. 1 and Section II.C.2 of Appendix H for Unit No. 2.

Alternate methods for compliance with Appendices G and H of 10 CFR Part 50 were proposed by the Pacific Gas and Electric Company and exemptions were requested from the identified requirements. Pacific Gas and Electric Company provided information in support of its method of compliance with Appendices G and H of 10 CFR Part 50 in its Final Safety Evaluation Report.

We have concluded from our review of the information submitted that exemptions to some of the specific requirements of Appendices G and H of 10 CFR Part 50 are required, and we have determined that the identified exemptions are justified. The bases for our determinations are discussed in the subsequent paragraphs of this section of the report.

The reactor vessels were manufactured for the Diablo Canyon site by Combustion Engineering, Inc., Chattanooga, Tennessee. The purchase orders were issued on April 23, 1968, for Unit No. 1 and on December 9, 1970, for Unit No. 2. Pursuant to paragraph 50.55a(c)(1) of 10 CFR Part 50, the ASME Boiler and Pressure Vessel Code in effect for Unit No. 1 is the 1965 Edition, including Winter 1966 Addenda and for Unit No. 2 is the 1968 Edition, including Summer 1968 Addendum. Since the ASME Code editions defined in 10 CFR 50.55a preceeded publication of Appendices G and H of 10 CFR Part 50, some of the fracture toughness tests for the ferritic materials in the primary coolant pressure boundary were not conducted to demonstrate explicit compliance with the current requirements in Appendices G and H.

Appendix G Exemptions

Based on our review of the applicant's submittal for compliance with Appendix G to 10 CFR 50 we have determined that the requirements of Appendix G have been met for Diablo Canyon Unit Nos. 1 and 2 except for Sections III.C.2 and IV.A.4. Section III.C.2 of Appendix G was not complied with to the extent that materials used to prepare test specimens for the reactor vessel beltline region were not taken directly from excess weld materials in the vessel shell courses following completion of the production longitudinal weld joint. The weld test specimens were taken from simulated weldments prepared from excess production plate, weld

wire and flux materials. However, in some instances the actual production weld wire and flux materials were not available, and the weld wire and flux materials used were duplicated to the extent practical in type and chemical composition. After weld preparation, the weldments were subjected to a heat treatment to obtain the metallurgical effects equivalent to those produced during fabrication of the reactor vessel.

Based on our evaluation of this information we conclude that an exemption from the specific requirements of Section III.C.2 of Appendix G is justified because the significant properties (e.g., weld wire chemical composition and weld flux type) of the weld materials in the beltline region will be representative of the actual beltline materials and their fracture toughness. The use of weldment test specimens having these simulated weld preparation and heat treatment conditions satisfy the intent of the specific requirement of Section III.C.2 of Appendix G and ensure an adequate margin of safety.

Section IV.A.4 of Appendix G was not complied with to the extent that Charpy V-notch tests were not conducted on the bolting materials to demonstrate a minimum toughness of 25 mils lateral expansion and 45 ft-lbs impact energy at the lower of either the preload temperature or at the lowest service temperature. An exemption is necessary because the measured absorbed energy on some of the test specimens was less than 45 ft-lbs. In addition, the lateral expansion was measured on only 10% of the test specimens.

We have reviewed the test data obtained on seven heats of bolting material used at the Diablo Canyon site, and find that 240 specimens were impact tested. The tests were conducted at 10°F. The average of all the impact energy values was 50.5 ft-lbs. The lateral expansion was measured on 24 specimens, and an average value of 35 mils was recorded. Referring to the fracture energy values obtained on the 240 specimens tested at 10°F, 90% of the values either met or exceeded the fracture toughness requirements of Appendix G of 10 CFR Part 50. The lowest value of 40 ft-lbs exceeded the special mechanical property requirements of paragraph N-330 of the 1965 Edition of the ASME Boiler and Pressure Vessel Code. Paragraph N-330 states that an average of 35 ft-lbs fracture energy is considered adequate for pressure vessel materials to be pressurized at ambient temperature (70°F).

Tests were performed at 10°F on 240 specimens taken from tubes and bars from 80 separate furnace charges. There were three specimens from each furnace charge. The average of all the impact energy values was 50.5 ft-lbs. The lowest average value from each heat was 45.7 ft-lbs and the highest average value was 61.0 ft-lbs. There were only five of the 80 furnace charges in which all three specimens failed to meet a minimum impact energy value of 45 ft-lbs. The lowest impact energy value on all the 240 specimens was 40 ft-lbs.

Based on our evaluation of these test data, we conclude that the exemptions for the areas of noncompliance to Appendix G of 10 CFR Part 50 are justified. Our conclusion is based upon the following:

Appendix G requires measurement of both lateral expansion and absorbed energy to provide additional assurance that the material has adequate fracture toughness. However, absorbed

impact energy and lateral expansion are very closely related criteria and provide an almost identical indication of material quality and toughness level. Consequently, we have determined that the measurement of the absorbed energy, in accordance with the ASME Boiler and Pressure Vessel Code requirements, is sufficient to demonstrate acceptable fracture toughness properties. As indicated above, a small percentage of the bolting materials tested at the ASME Code 10°F test temperature had absorbed energies less than the 45 ft-lbs required by Appendix G. While the Appendix G requirements were not met exactly in these instances, the available test data are sufficient to indicate that these materials were manufactured properly, are of acceptable quality and have adequate fracture toughness to provide reasonable assurance that adequate safety margins will be obtained and maintained during operation. Further, conducting the impact tests at 10°F, instead of a higher preload or service temperature, as required by Appendix G, is conservative because the absorbed energy normally increases with increasing temperature.

We have evaluated the data presented in the FSAR and based on the results of our evaluation we have determined that while the precise requirements of Appendix G have not been met, sufficient information has been provided to demonstrate that the safety objective of Appendix G has been achieved.

Appendix H Exemptions

Based on our review of the applicant's submittal for compliance with Appendix H to 10 CFR 50 we have determined that the requirements of Appnedix H have been met except for Section II.B for Unit No. 1 and Section II.C.2 for Unit No. 2.

Section II.B of Appendix H of 10 CFR Part 50 requires that the fracture toughness properties of the beltline region of reactor vessels be monitored by a surveillance program complying with the American Society for Testing and Materials (ASTM) Standard E 185-73, "Surveillance Tests for Nuclear Reactor Vessels." The surveillance test program for the Diablo Canyon site, Unit No. 1, complies with ASTM E 185-73 except that the orientation of the base metal specimens and the number of specimens per capsule complies with ASTM E 185-70. The test materials required by ASTM E 185-73 are orientated transverse to the major rolling direction and consist of 12 Charpy V-notch impact specimens from the base metal, heat affected zone, and the weld metal and two tensile specimens from the base and weld metals. A total of five capsules containing these materials are required. Transverse test materials are not required by ASTM E 185-70.

The surveillance program for the Diablo Canyon site, Unit No. 1, contain Charpy V-notch and tensile test specimens from the limiting base metal and representative weld metal and heat affected zone metal from the beltline region of the reactor vessel. In addition, they contain correlation monitors of documented specimens of SA 533 Grade B Class 1 pressure vessel steel obtained through ASTM Committee E-10, Radioisotopes and Radiation Effects Committee, and wedge-opening-loading (WOL) type specimens. These additional specimens will aid in evaluating the fracture toughness properties of materials in the beltline region throughout the life of the reactor vessel and provide information beyond that required by compliance to ASTM Standard E 185-73.

We conclude from our evaluation of the surveillance program for the Diablo Canyon site, Unit No. 1, that it complies with the intended purpose of Appendix H of 10 CFR Part 50. The number of Charpy impact and tensile specimens and the withdrawal schedule during reactor operation are adequate to evaluate the shift in the nil- ductility temperature and the decrease in the upper shelf fracture energy of reactor beltline materials. Further the WOL specimens provide additional assurance that the facture toughness of the reactor vessel beltline material will be adequately monitored during plant operation. Although the orientation of the impact specimens is not in compliance with ASTM 185-73, the staff has acceptable methods for estimating the transverse impact properties from data obtained on longitudinally orientated specimens. Further, if necessary, the degradation of the beltline materials as a result of neutron irradiation can be estimated independently to ensure safe reactor operation by following the recommendations of Regulatory Guide 1.99, "Effects of Residual Elements on Predicting Radiation Damage to Reactor Vessel Materials."

Based on our evaluation of the information provided by the applicant, an exemption for Unit 1 to Section II.B of Appendix H of 10 CFR Part 50 is justified. There is reasonable assurance that the surveillance program for Unit No. 1 of the Diablo Canyon site will monitor the change in the fracture toughness properties of the limiting materials in the beltline region to a degree adequate to determine the temperature-pressure limits in order to preserve the integrity of the vessel. The program will generate sufficient information to permit the determination of conditions under which the reactor vessel will be operated with an adequate margin of safety against rapidly propagating fracture throughout its service lifetime.

Section II.C.2 of Appendix H was not complied with for Unit No. 2 to the extent that four of the six surveillance capsules will receive a neutron flux exceeding three times that received by the inner surface of the reactor vessel. The purpose of this requirement in Appendix H is to restrict the location of the surveillance specimens and to ensure that the surveillance specimens receive similar neutron fluences as those received by the inner surface of the reactor vessel.

We have reviewed the fracture toughness test data obtained from operating reactor surveillance programs and conclude that no significant inaccuracy will result from the slight increase in lead factor. Further, techniques and recommendations are available in Regulatory Guide 1.99, "Effect of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials" to estimate the radiation damage in reactor vessel materials, including relatively small inaccuracies occurring from lead factors greater than three and ensure an adequate margin of safety against rapidly propagating fracture.

Based on our review and evaluation of operating data and data that will be generated from the surveillance program, we conclude that an exemption for Unit 2 from Section II.C.2 of Appendix H to 10 CFR Part 50 is justified because experience has proved the reliability of techniques for estimating radiation damage as a function of fluence is independent of small variations in the neutron lead factor.

Our technical evaluation has not identified any practical method by which the existing Diablo Canyon site, Unit Nos. 1 and 2, reactor vessel can comply with the specific requirements of Appendices G and H, 10 CFR Part 50. Requiring compliance with the identified specific requirements would delay the startup of the site due to the need to complete the following actions: (1) obtain specimens from the beltline region of the reactor vessels from the weld metal and heat affected zone, (2) retest the bolting materials to confirm compliance with Appendix G, and (3) relocate the installed material surveillance specimens.

Findings on Requested Exemptions to Appendices G and H

Based on the foregoing, pursuant to 10 CFR Section 50.12, we find that the requested exemptions to the specific requirements of Appendices G and H of 10 CFR Part 50 as discussed above are authorized by law and can be granted without endangering life or property or the common defense and security and are otherwise in the public interest. We conclude that the public is served by not imposing certain provisions of Appendices G and H of 10 CFR Part 50 that have been determined to be either impractical or would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

Furthermore, we have determined that the granting of these exemptions do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. We have concluded that these exemptions would be insignificant from the standpoint of environmental impact and pursuant to 10 CFR 51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal, need not be prepared in connection with this action.

Operating Limitations

The reactor will be operated in a manner that will minimize the possibility of rapidly propagating failure. The pressure- temperature limits will be established by the methods identified and recommended by NRC Standard Review Plan 5.3.2, Appendix G of Section III of the ASME Boiler and Pressure Vessel Code, and Appendix G of 10 CFR Part 50.

Additional conservatism in the operating limits used for heatup, cooldown, testing and core operation will be provided by programming the plant computer to continually compare the actual operating limits to a pressure-temperature curve more restrictive than that required by Appendix G of 10 CFR Part 50. Conservatism in the pressure-temperature limits will be provided because they will be determined assuming that the beltline regions of the reactor vessels have already been irradiated.

The use of NRC recommendations, Appendix G of the ASME Boiler and Pressure Vessel Code, and Appendix G of 10 CFR Part 50 to establish safe operating limitations for the reactor vessels will ensure adequate safety margins during operating, testing, and maintenance and anticipated operational transients. Compliance with these ASME Code provisions and Commission regulations constitutes an acceptable basis for satisfying the requirements of Criterion 31 of the General Design Criteria.

Reactor Vessel Material Surveillance Program

The toughness properties of the reactor vessel beltline material will be monitored throughout the service life with a material surveillance program that meets the requirements of the American Society for Testing Materials Specification (ASTM) E 185, "Surveillance Tests for Nuclear Reactor Vessels." The evaluation of the irradiation damage to the Diablo Canyon reactor vessels will be assessed on pre-irradiation and post-irradiation testing of charpy V-notch, tensile, dropweight and wedge-opening-loading specimens. The specimens will be irradiated in capsules located near the core midheight and removed at specified intervals.

The removal schedule for the surveillance specimens conform to the requirements of ASTM E 185-73 and Appendix H of 10 CFR Part 50 for Unit No. 1. The specimen orientation and number of specimens per capsule conform to ASTM E 185-70, which was in effect at the time the reactor vessel was manufactured. Weld metal specimens are contained in three of the eight capsules.

The removal schedule, specimen orientation, number and type of specimens per capsule conform to ASTM E 185-73 and Appendix H of 10 CFR Part 50 for Unit No. 2. However, four of the six capsules lead the reactor vessel fluence by a factor of 3.6.

We conclude that changes in the fracture toughness of the Diablo Canyon reactor vessel beltline materials caused by exposure to neutron irradiation will be properly assessed and that adequate safety margins against the possibility of vessel failure will be provided since the material surveillance requirements of ASTM E 185 and Appendix H of 10 CFR Part 50 are met. Compliance with these specifications, recommendations, and NRC regulations ensures that the surveillance program constitutes an acceptable basis for monitoring radiation induced changes in the fracture toughness of the reactor vessel material, and satisfies the requirements of the Commission's General Design Criterion 31.

5.2.7 Steam Generator Material

5.2.7.1 Secondary Water Chemistry

In late 1975 we incorporated provisions into the Standard Technical Specifications (STS) that required limiting conditions for operation and surveillance requirements for secondary water chemistry parameters. The Technical Specifications for all pressurized water reactor plants that have been issued an operating license since 1974 contain either these provisions or a requirement to establish these provisions after baseline chemistry conditions have been determined. The intent of the provisions was to provide added assurance that the operators of newly licensed plants would properly monitor and control secondary water chemistry to limit corrosion of steam generator tubes and the tube support plates.

In a number of instances, the Technical Specifications have significantly restricted the operational flexibility of some plants with little or no benefit with regard to limiting degradation of steam generator tubes. Based on this experience and the knowledge gained in recent years, we have concluded that Technical Specification limits are not the most effective way of assuring that steam generator tube degradation will be minimized.

Due to the complexity of the corrosion phenomena involved and the state-of-the-art as it exists today, we believe that, in lieu of specifying limiting conditions in the Technical Specifications, a more effective approach would be to institute a license condition that requires the implementation of a secondary water chemistry monitoring and control program containing appropriate procedures and administrative controls.

The required program and procedures are to be developed by the applicant with input from its reactor vendor or other consultants, to more readily account for site and plant-specific factors that affect chemistry conditions in the steam generators. In our view, plant operation following such procedures would provide assurance that licensees would devote proper attention to controlling secondary water chemistry, while also providing the needed flexibility to allow them to deal more effectively with any off-normal conditions that might arise.

Consequently, we requested, in a letter dated August 2, 1979, that the applicant propose a secondary water chemistry program which will be referenced in a condition to the license. In the letter we concluded that such a license condition, in conjunction with existing Technical Specifications on steam generator tube leakage and inservice inspection, would provide the most practical and comprehensive means of assuring that steam generator tube integrity would be maintained.

In a letter dated April 8, 1980, the applicant committed to implement a water chemistry monitoring and control program for the Diablo Canyon Nuclear Plant which will include the following:

1. Identification of a sampling schedule for the critical parameters and of control points for these parameters;

2. Identification of the procedures used to measure the value of the critical parameters;

3. Identification of process sampling points;

4. Procedure for the recording and management of data;

5. Procedures defining corrective actions for off-control point chemistry conditions; and

6. Procedures identifying (a) the authority responsible for the interpretation of the data and (b) the sequence and timing of administrative events required to initiate corrective action.

We have reviewed the applicant's program and conclude that it meets our requirements as delineated in our letter of August 2, 1979.

The proposed secondary water chemistry program will sample and monitor the effluent of the condensate of the main steam condenser, however, when condenser leakage is confirmed, we will require the applicant to repair or plug the leak in accordance with Branch Technical

Position MTEB 5-3 appended to Standard Review Plan 5.4.2.1. The license will be conditioned accordingly.

It should be noted that the steam generators of Diablo Canyon Nuclear Plant, Units 1 and 2 are of the Westinghouse "51" series design having carbon steel supporting plates with drilled tube support holes. Steam generators of this design in operating plants have experienced denting and cracking. Although an effective secondary water chemistry control program can reduce the rate of tube degradation there is no assurance that a 40-year steam generator lifetime can be obtained.

In spite of the possibility of tube cracking, we have concluded that operation of the steam generators will not constitute an undue risk to the health and safety of the public for the following reasons:

(1) Primary to secondary leakage rate limits and associated surveillance requirements have been established to provide assurance that the occurrence of tube cracking during operation will be detected and that appropriate corrective action, such as tube plugging, will be taken such that any individual crack present will not become unstable under normal operating, transient, or accident conditions.

(2) Inservice inspection requirements and preventative tube plugging criteria have been established to provide assurance that the great majority of degraded tubes will be identified and removed from service before leakage develops.

5.2.7.2 Steam Generator Inspection Ports

In our letter of May 8, 1980, we requested that the Pacific Gas and Electric Company install inspection ports in the steam generators of the Diablo Canyon Nuclear Plant, Units 1 and 2 prior to the start of the operations after the first refueling. These ports were to facilitate monitoring the progression of tube denting and tube support plate degradation and to facilitate the removal of tube sections for laboratory examinations.

For some forms of steam generator degradation which have occurred, eddy current testing and tube gauging alone are not sufficient to assess and monitor tube support plate degradation. In order to perform adequate assessment and monitoring of these areas, it is necessary to install inspection ports. These ports should be installed just above the upper support plate and between the tubesheet and the lower support plate and in line with the tube lane.

Under the As Low As Reasonably Achievable (ALARA) concept, we have been requesting that all possible steam generator modifications be made before the start of operations in order to minimize personnel exposure. Based upon experience at Surry 1, the ports can be installed in three steam generators at a total personnel exposure of 7.5 man-rem. On this basis, although installation prior to initial operation is preferable, we have determined that the potential installation exposure following the first cycle of operation is not significant enough to justify the delay of the initial start-up of the plant to permit the installation of inspection ports.

However, since secondary side contamination will increase as the operating time increases, we require that these ports be installed prior to start-up after the first refueling. Accordingly, in the event that the ports have not been installed, before the license has been issued the license of each unit will reflect this requirement.

5.2.7.3 Row 1 Steam Generator Tubes

Experience has shown that the small bend radius of the Row 1 tubes in the steam generators of Westinghouse design leads to early onset of cracking. At the present time, Westinghouse has committed (letter from T. M. Anderson to R. H. Vollmer, May 12, 1980) to a program to determine the particular susceptibility of Row 1 tubes to cracking. The program involves removing numerous tubes from the Trojan plant and subjecting them to non-destructive and destructive testing to identify the cause of the cracking and to develop a field inspection method capable of detecting potential leaking tubes. The results of this evaluation are expected to be available in October 1980; thus a sound engineering decision on the need to plug Row 1 tubes can be made prior to the issuance of the full power license. We shall review the program results and decide at that time on the necessity to plug the Row 1 tubes.

5.2.7.4 Control Rod Guide Tube Support Pins

On March 13, 1980, Westinghouse reported to the NRC that Inconel 750 control rod guide support pins that were given a low temperature solution heat treatment may be susceptible to stress corrosion cracking. This followed recent support pin inspections at a foreign plant which revealed stress corrosion cracks in Westinghouse-supplied pins. We believe that the existing guide support pins in the Diable Canyon Nuclear Plant Units Nos. 1 and 2 should be replaced with new pins that have been given a heat treatment which produces a condition that is highly resistant to stress corrosion cracking. The applicant has completed the above cited pin replacement for Unit 1 and has committed to replace the pins on Unit 2; therefore, we consider this matter resolved.

6.0 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 Containment Response to Main Steam Line Break Accidents

In Section 6.2.1 of SER Supplement 6, we made the following statement:

> "Our recent evaluations of a postulated main steam line break inside containment have indicated a potential concern in two areas: (1) The peak calculated containment pressure and temperature, and (2) the environmental qualification of safety-related equipment located inside containment that must function."

We also stated that we would review further information to be submitted by the applicant and provide our evaluation in a future supplement to the SER.

The applicant has analyzed a spectrum of main steam line break (MSLB) accidents, considering various single active failures, to determine the containment peak pressure profile and peak temperature profile. These are used in determining the environmental qualification conditions for safety-related equipment.

Mass and energy releases for a spectrum of steam line breaks were calculated using the MARVEL code described in Topical Report WCAP-8860, "Mass and Energy Releases Following a Steam Line Rupture."

The MARVEL code describes the primary and secondary systems of a PWR including the power excursion which may occur in the core following a main steam line break. The code calculates heat flow from the core and intact steam generators into the primary system, and heat flow from the primary system into the broken steam line. The primary system heat flow produces additional steam which is added to the containment. No liquid entrainment is assumed to flow from the break so that the break flow is all steam. This assumption permits the secondary liquid to remain in the steam generator until it is boiled by heat flow from the primary system, and maximizes the energy release. Using this assumption for a double-ended break, 55×10^6 additional BTUs are added to the containment relative to the case with entrainment. This additional energy provides a margin of conservatism in the analysis. The analysis includes additional steam from the intact steam generators before closure of the isolation valves and the unisolated steam in the steam lines and turbine plant piping. Feedwater flow is added to the affected steam generator based on runout flow before isolation. No credit is taken for any feedwater flow reduction during the valve closure period. The unisolated feedwater mass is added to the steam generator inventory during the blowdown. We have concluded that the mass and energy release data are conservative for containment analysis WCAP-8860, which describes MARVEL and its application for mass and energy release calculations,

also provides for the assumption of entrained liquid leaving the break which acts to reduce the containment pressure and temperature. The model with entrainment is currently under review by the NRC. Our review at this time indicates that there is reasonable assurance that the mass and energy release rates will not be appreciably altered by completion of the analytical review.

The applicant performed detailed analyses for a total of 50 postulated pipe breaks, comprising a spectrum of break areas, power levels, and single active failures. Most of these analyses were done with the Westinghouse COCO code model developed for the IEEE-323-1971 Westinghouse Supplemented Equipment Qualification Program. Two cases were done with the COCO code modified to be consistent with the staff's interim containment model. These two cases were for the worst temperature and worst pressure cases, and were redone for comparison between the Westinghouse and staff containment models.

The case resulting in the highest calculated peak pressure (44.9 psig) was the 3.69 ft^2 break occurring while at 70 percent power.

The case resulting in the highest calculated peak temperature (333°F) was for the 0.910 ft^2 break occurring while at 70 percent power. When the applicant re-analyzed this case with the COCO code, modified to be consistent with the staff's interim containment model, the resulting peak temperature was 344°F (and pressure of 41.0 psig). We have performed a confirmatory analysis for this case with our CONTEMPT-LT/26 containment transient code, and our calculated peak temperature and pressure agree quite closely with the 344°F and 41.0 psig results of the applicant.

Therefore, we conclude that the applicant's analysis, using the staff's interim containment model, is acceptable, and that the containment atmosphere temperature and pressure profiles with peaks of 344°F and 41.0 psig are acceptable for use in the environmental qualification of safety-related qualification of safety- related equipment located inside containment. The matter of environmental qualification of safety-related equipment located inside containment that must function during an accident is addressed in Section 7.8 of this Supplement to the SER.

6.2.3 Containment Air Purification and Cleanup Systems

In Section 6.2.3 of SER Supplement 7, we stated that we would reevaluate the containment purging system for conformance with our Branch Technical Position (BTP) CSB 6-4, "Containment Purging During Normal Plant Operations," in the Standard Review Plan Section 6.2.4 which describes design provisions and analytical methods that are acceptable for providing assurance that this system will not significantly increase the calculated doses due to a loss-of-coolant accident if the system is in use when an accident occurs including the capability of the purge valves to close when exposed to a loss-of-coolant environment and potential debris. The applicant has addressed BTP CSB 6-4.

As recommended in the Branch Technical Position, the applicant has provided systems within the reactor containment building to control the temperature in the containment and filter the containment air to reduce the airborne activity in the containment. Thus, the frequency and duration of containment purging to permit personnel access will be reduced.

The purge system's containment isolation valves are signalled to close by either high containment airborne radioactivity levels or by a safety injection signal. One of the input parameters that initiates a safety injection signal is high internal containment pressure (3 pounds per square inch gauge). We, therefore, find that there is acceptable diversity in the parameters sensed to initiated valve closure.

The purge system consists of one 48-inch purge line and one 48-inch vent line; in addition, there is a 12-inch containment atmosphere vacuum/overpressure relief line. Since the applicant's off-site dose analysis was performed assuming the 12-inch vacuum/overpressure valve closed, the applicant has committed to not using the purge/vent lines when the vacuum/overpressure line is being used. The plant's Technical Specifications will include this requirement.

We have performed a dose consequence analysis for an assumed loss-of-coolant accident while the containment is being purged. In performing the analysis we have assumed the purge and vent lines both to be open. A pre-existing iodine spike in the reactor coolant system fluid and three second valve closure times (includes a one second delay for time necessary to generate a containment isolation signal) were also assumed. The results of our analysis show that in the event of a loss-of-coolant accident during purge system operation, site boundary doses would not exceed the dose guidelines of 10 CFR Part 100, assuming valve operability explained below. Valve closure times have been determined by test to be approximately two seconds, which is within the criteria assumed in our dose consequence analysis.

The applicant has provided debris screens inboard of the inside containment isolation valves, to assure that the 48-inch purge system containment isolation valves will not be prevented from closing by debris following a loss-of-coolant accident. The applicant has concluded on the basis of analyses that the 48-inch purge system containment isolation valves would properly close if a loss-of-coolant accident occurred when these valves were open. We have reviewed the applicant's analyses and conclude that the 48-inch containment purge valves will operate when required during a loss-of-coolant accident. The applicant has committed to limit purging operations to no more than 90 hours per year. The applicant has also committed to blocking the 12-inch vacuum/overpressure relief line valves to no more than 50 degrees of full open until the applicant has submitted analyses demonstrating operability of these valves. Thus, the 90-hour limit is a combined total for use of either the purge system or the vacuum/overpressure relief system. This will limit the probability of a LOCA event occurring while purging. We find these interim measures for initial operation to be acceptable until the question of valve operability is resolved. We will report on the resolution of this issue in a future supplement to the Safety Evaluation Report for full power operation.

The applicant has provided the necessary test connections to permit leak testing of the containment isolation valves in the purge system piping. We require that the containment isolation valves be local (type C) leak rate tested following each use of the system. This is consistent with the action taken regarding leak testing of the purge system containment isolation valves for the Donald C. Cook Nuclear Plant, Unit 2, McGuire Nuclear Station, Units 1 and 2 and the Sequoyah Nuclear Plant, Units 1 and 2. This requirement will be included in the Technical Specifications.

We therefore conclude that the containment purge system and the vacuum/overpressure relief system may be used, under the restrictions detailed above, during the normal plant operating modes of startup, power, hot standby, and hot shutdown, but both systems may not be used at the same time, as stated above.

6.2.6 Containment Leakage Testing Program

The applicant has provided information describing its method of compliance with 10 CFR Part 50, Appendix J, relating to leak testing of the containment airlocks after each opening. As a result of our review of this information, we have determined that an exemption to 10 CFR Part 50, Appendix J is required and justified. Our bases for this conclusion are discussed below.

Appendix J Exemption

Paragraph III.D.2 of Appendix J to 10 CFR Part 50 requires in part that containment airlocks be tested at six-month intervals and after each opening. Paragraph III.B.2 of Appendix J to 10 CFR Part 50 requires that these tests be performed at a pressure not less than the calculated peak containment internal pressure related to the design basis accident (47 pounds per square inch gauge). The airlock design of Diablo Canyon, Units 1 & 2 includes dual seals on the airlock doors with the capability to apply a pressure between the seals. This will permit door seal integrity to be demonstrated without pressurizing the entire airlock.

Findings on Requested Appendix J Exemptions

Based on plant operating experience, we find that the leakage testing of containment airlocks after each opening as required by Paragraph III.D.2 of Appendix J to 10 CFR Part 50, when frequent airlock usage is necessitated over a short period of time is, in our judgment, impractical and unnecessary to assure the maintenance of the leaktight integrity of the airlocks. It is our judgment that verification of the leak-tightness of the airlock by leak testing of the door seals within 72 hours after being opened, at a reduced pressure (10 pounds per square inch gauge), provides the required assurance that the leaktight integrity of the airlock is maintained. The effect on accident consequences of testing after each opening versus testing within 72 hours of an opening is judged to be insignificant. Furthermore, if an airlock door seal is damaged, it will be manifested during testing at the lower pressure of 10 pounds per square inch gauge. The Diablo Canyon Technical Specifications will provide that the airlocks be tested at six month intervals at the test pressure of 47 pounds per square inch gauge as required by Paragraphs III.D.2 and III.B.2 of Appendix J to

10 CFR Part 50 and the airlock door seals be leak tested within 72 hours after being opened. These six-month tests involve pressurization of the entire airlock instead of just the gap between the door seals.

Based on the foregoing, pursuant to 10 CFR Section 50.12 we have determined that an exemption to the specific requirements of Appendix J to 10 CFR Part 50 as discussed above is authorized by law and can be granted without endangering life or property or the common defense and security and is otherwise in the public interest. We conclude that the public is served by not imposing certain provisions of Appendix J to 10 CFR Part 50 that have been determined to be either impractical or would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

We therefore conclude that the methods for leakage testing of the containment airlocks as provided for in the Diablo Canyon Technical Specifications represent an acceptable alternative to those required by Appendix J to 10 CFR Part 50. Accordingly, the proposed exemption from those requirements is justified and will be required prior to the issuance of an operating license.

6.3 Emergency Core Cooling System (ECCS) Metal Water Reactor Heat Rates

As noted in SER Supplement 7, Westinghouse informed the staff that it had discovered an error in the zirconium-water reaction logic of its evaluation model which had the effect of causing the metal-water reaction heat release to be one-half of what it should be.

To account for the error the applicant has provided reanalyses performed using a corrected and approved evaluation model, and using parameters for Diablo Canyon, Units 1 and 2, that compensate for the error results including containment backpressure which has been reviewed and approved by the staff.

We find that previously reviewed and approved analyses are acceptable to identify the worst type and location of break to be a large double-ended cold leg guillotine (DECLG) rupture for both units.

The applicant has provided additional analyses to determine the worst case discharge coefficient (C_d) and worst case results for each unit.

For Diablo Canyon Unit 1, Salem Unit 2, has been found to be acceptable as reference to identify the worst case C_d. Salem reanalyses, which were reviewed and approved have identified the worst case C_d to be 0.8. The DECLG, C_d=0.8, case was reanalyzed for Diablo Canyon, Unit 1. The calculated peak clad temperature is 1930°F which is below the acceptable limit (2200°F) specified in 10 CFR 50.46. In addition, the calculated maximum local metal/ water reaction of 3.0 percent and a total core-wide metal/water reaction of less than 0.3 percent are well below the allowable limits (10 CFR 50.46) of 17 percent and 1 percent, respectively.

Because of Unit 1 and Unit 2 vessel internals differences, which could possibly have influenced the spectrum of C_d breaks, a three-break reanalysis was performed for Diablo Canyon, Unit 2. This reanalysis determined that the worst case C_d for Diablo Canyon, Unit 2 is 0.8 also. For this case, the calculated peak clad temperature is 2187°F which is below the acceptable limit (2200°F) specified in 10 CFR 50.46. In addition, the calculated maximum local metal/water reaction of 7.5 percent and a total core-wide metal/water reaction of less than 0.3 percent are well below the allowable limits (10 CFR 50.46) of 17 percent and 1 percent, respectively.

The containment parameters used to determine the backpressure for the LOCA analysis are the same as those used previously and found acceptable. The single failure evaluation of the ECCS was previously reviewed and found acceptable. The applicant has stated that the reactor core and internals have been designed so that the reactor can be safely shut down and the essential heat transfer geometry of the core preserved following a postulated LOCA. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling and the cladding oxidation limits of 17% are not exceeded during or after quenching.

The analysis was performed with a total peaking factor of 2.32 at a power of 102% of 3411 MWt for Unit 2 and 102% of 3338 MWt for Unit 1. Based on this review and that in the FSAR Amendments 47 and 49, the staff concludes that the reanalyzed ECCS cooling performance, corrected for the zirconium-water error, is acceptable and conforms to the criteria in 10 CFR 50.46.

6.3.2 System Design

Refueling Water Storage Tank (RWST) Capacity

Subsequent to Diablo Canyon SER Supplement No. 8, the staff expressed generic concerns about refueling water storage tank capacity in a board notification memorandum (R. Tedesco to D. Vassallo, Re: Seabrook Station, Docket No. 50-443/444, February 15, 1979). We requested that the applicant address the concerns of that board notification.

In submittals to the staff on March 20, 1980 and April 16, 1980, the applicant has provided analyses to demonstrate the adequacy of the RWST capacity to supply ECCS pumps with water during a LOCA until completion of the switchover from the injection mode to the recirculation mode.

In the submittals, the applicant indicated that this Technical Specification should require a minimum RWST inventory of 400,000 gallons of water. Analyses based upon containment sump geometry indicate that 205,000 gallons of injected water will flood the sump to an elevation of 92 feet, 9 inches which will provide adequate NPSH for ECCS pumps. To accommodate this water volume, there will be a low level alarm when 176,000 gallons of inventory remain to alert the operator to initiate the switchover procedure. A protective automatic RHR pump trip will also be provided at 176,000 gallons. These two measures allow sufficient water inventory (224,000 gallons) to be injected prior to switchover to assure sump flooding to the elevation of 92 feet, 9 inches.

After the low level alarm, the applicant has calculated that 70,500 gallons of usable RWST water are needed to complete switchover for the case of all pumps operating. Considering instrument error and usable volume (due to the elevation of the ECCS pump suction piping inlets), at least 103,000 gallons are available in the RWST to complete the switchover and to continue supplying containment spray pumps for pH considerations in the sump.

Based on the above analyses and associated Technical Specifications, alarms, controls, and procedures, we find that the proposed RWST will have an acceptable water supply.

Automatic Switchover

In particular, we reviewed the applicant's analysis of available time for operator response and the switchover actions required and conclude that the proposed manual procedure with supplemental automatic pump trip is acceptable; however, we have informed the applicant by letter that during the longer term we require that the switchover procedure be more fully automated to reduce the needed operator actions subsequent to a LOCA.

6.4 Habitability Systems

In our SER of October 1974, we concluded that the appropriate chlorine protection devices proposed by the applicant for protection against the effects of accidental release from onsite sources of chlorine were acceptable. Subsequently, in Amendment 81 to the FSAR, the applicant described extensive changes to the control room ventilation system. This supplement contains our evaluation of the revised system in regard to the impact of an accidental release of chlorine upon control room habitability.

The applicant has provided additional fans, controls, and ducts to provide for pressurization of the control room with air from a remote intake in the event of a nuclear accident. Since our previous analysis had indicated that the radiological protection provided the operator for such an accident with the intake located near the containment was within the guidelines of GDC 19, and therefore acceptable, the new system, which provides additional protection, is clearly also acceptable.

These changes also affected the protection provided against accidental chlorine releases. In addition to chlorine detectors in the normal air intakes for the control room, additional redundant detectors are located in each remote intake. All the new equipment is designed to conform to seismic Category I, and is protected from external missiles. The detectors will automatically initiate isolation of the control room, with alarm, and will permit a maximum of 0.013 pounds of chlorine through the intake before the dampers are closed. An additional 0.002 pound would enter by infiltration under these conditions. When this is diluted uniformly in the control room air volume of 170,000 cubic feet, the resulting concentration is 0.44 ppm, well within the 15 ppm limit for a 2 minute exposure given in Regulatory Guide 1.78.

Our conservative, independent analysis indicates that the resulting concentrations in the control room are within the toxicity limits given in Reg. Guide 1.78 and are, therefore,

acceptable. Hence we conclude that the design of the control room including its ventilation systems and controls provide an acceptable degree of protection to the operators from the hazards of the chlorine to be stored at this site.

7.0 INSTRUMENTATION AND CONTROL

7.6 RHR Isolation Valve Position Indication

In SER Supplement 8 we postulated that fire damage to electrical cables could cause both RHR suction line isolation valves to open. The applicant has since committed to remove power from the valve motor operators to prevent this undesired operation. We find this response acceptable.

7.8 Environmental and Seismic Qualification

Environmental Qualification of Submerged Equipment

We stated in SER Supplement 7 that we had identified the RCS Pressure, Pressurizer Level, and Narrow Range Steam Generator Level Transmitters as being vulnerable to malfunction due to submergence and that the applicant had agreed to replace them with qualified instruments or to justify their continued use. We have reviewed additional information that presents the results of submergence testing of the replacement Barton transmitters. This information demonstrates that these transmitters will perform satisfactorily when submerged. We find this response acceptable.

Environmental Qualification - Nuclear Steam Supply System Equipment Exposed to Normal Environments

Supplement 7 to the SER stated that we would complete our evaluation of temperature ranges and of the ambient temperature monitoring system in a future supplement. The ambient temperature monitoring system that monitors the areas containing BOP safety-related equipment will also monitor areas outside of containment that house NSSS safety-related equipment which exceed, with adequate margin, the design basis controlled temperature range of the environmental control systems. Should the ambient temperature of given areas exceed this design basis controlled temperature range, there would be ample time to evaluate the affected safety equipment's capability for continued service and take corrective action, if required. We find this acceptable.

Environmental Qualification - Class IE Equipment Exposed to Severe Environments

In SER Supplement 7 we required the applicant to furnish a listing of all BOP and NSSS safety-related equipment that may be required to function under severe environmental conditions. This list was to identify the equipment, its manufacturer, its model number, its location, and a specific reference to its qualification report. We reviewed FSAR Table 3.11-1A which lists this equipment and provides the requested information. We find this response acceptable.

(a) We questioned the adequacy of the qualification of Rosemount pressure and differential pressure transmitters to survive the extreme environmental conditions produced by high energy line breaks inside containment. Based on our reveiw of the qualification report for these transmitters, we conclude that a sufficient basis was not provided to justify their use throughout the life of the plant. Since the test conditions to which these transmitters were subjected did not result in a failure of the transmitter to respond to changes in measured process conditions, we find that they are acceptable for use in the interim. Accordingly, we will condition the operating license to permit the use of Rosemount pressure and differential pressure transmitters until the second refueling outage. At this time, requalification of these transmitters or replacement transmitters that have been qualified will be required.

(b) We reviewed Westinghouse Topical Report WCAP-9157 which contains the environmental qualification results for the main coolant loop resistance temperature detectors (RTD). These temperature sensors provided data to confirm natural circulation cooling as well as data to ensure an adequate margin of subcooling to prevent steam formation in the reactor coolant system. We questioned the basis for the assessment that the normal and post accident radiation exposure would be limited to a radiation dose for which the RTDs were qualified. The applicant provided a response to our concern which concluded that the RTDs used for post accident monitoring are adequate if replaced after 14 years of operation. We conclude that this evaluation did not include assumptions which contained an adequate degree of conservatism. Therefore, we will condition the operating license to require the replacement of RTDs used for post accident monitoring at each refueling outage pending requalification of the sensor to a higher radiation dose which is established based on a conservative assessment of post accident radiation levels and the normal radiation dose for their service life.

Westinghouse Topical Report WCAP-9157 also included the qualification of a containment pressure measuring system consisting of a Barton pressure transmitter located outside containment connected to a remote bellows located inside containment. The environmental tests of the remote bellows assembly produced a pressure spike that caused an error in the pressure measurement which exceeded the acceptance criteria for instrument error. Following further investigation of the anomalous indication, Westinghouse has determined that flashing of the water used in this filled system sensor is the cause of this problem. Tests have further demonstrated that this phenomena does not occur with the use of fill fluids which will not flash when subjected to the required environmental conditions. Westinghouse has proposed to requalify the remote bellows sensor using an appropriate radiation resistant fill fluid. PG&E has committed to replace the fill fluid in this system prior to power operation to resolve this concern. We conclude that this action is adequate to satisfy the qualification requirements for the containment pressure measurement system.

(c) In September 1978, Westinghouse provided test results for the environmental qualification of Barton Models 763 and 764 Lot 1 transmitters (Letter Report NS-TMA-1950). Our conclusion, based on these tests, was that the instruments would perform their short-term safety functions. However, we required that additional testing be conducted to

confirm their capability for longer term post-accident monitoring. In September 1979, Westinghouse provided the results of these supplemental tests.

In the original tests, it was attempted to demonstrate the qualification of these transmitters by subjecting them to high ratiation levels corresponding to post-LOCA conditions and subsequently exposing them to the high temperature steam conditions, typical of main steam line break (MSLB) accidents. This combined test was performed to circumvent the need for separate LOCA and MSLB tests. This combination of high radiation and temperature while not causing the transmitters to fail, resulted in excessive instrument error.

The supplemental tests which followed were based upon radiation levels and subsequent exposure to a steam environment corresponding to LOCA and MSLB conditions separately. Additional tests were also conducted to investigate the effects of radiation and temperature separately and in combination. This was done to promote an understanding of the phenomena which caused the errors and to provide a basis to support the conclusion that the transmitters are qualified to operate satisfactorily under the required service conditions. While the supplemental tests results support the conclusions that the Lot 1 instruments will function in an accident environment, we do not believe that these instruments provide a sufficient margin of safety to justify their use throughout the life of the plant. Further improvements to obtain an additional margin of safety are warranted due to the safety significance of the information provided for post-accident recovery by these instruments. Accordingly, we will condition the operating licenses to permit the use of the Lot 1 Barton Transmitters until the second refueling outage. At that time, modified or replacement transmitters, that have been demonstrated to have a greater tolerance to harsh environments, will be required.

(d) In June of 1979, Westinghouse reported a potential safety hazard under 10 CFR 21. This report addressed errors caused in steam generator level indication following high energy pipe breaks inside containment. High ambient temperatures due to accidents can result in a decrease in the water column density for the level instrument reference leg with a consequent increase in the indicated steam generator water level (i.e., indicated water level exceeding actual level). We requested that PG&E evaluate the effects of such errors for all level measurement systems in containment. This review led to a decision to insulate the reference legs for steam generator level measurements.

The low-low steam generator level trip setpoint is adjusted above zero measured level by an amount which just equals the accumulation of all system errors, including temperature effects on the reference legs. We do not find this approach to evaluating errors and establishing the setpoint for safety action to be acceptable. The choice of zero measured level, as a reference to establish the setpoint, does not provide an adequate margin of safety since these level transmitters do not respond to a reduction of water level below this point in the steam generators. Accordingly, we will condition the license to require a minimum low-low steam generator level setpoint of 16 percent (a margin of 3 percent in addition to identified errors of 13 percent) until such time as it can be demonstrated that an adequate margin of safety exists within the present analysis.

We have reviewed the applicant's evaluation of level measurement errors and their impact on post-accident operation to assure that adequate water levels will be maintained in the pressurizer and steam generator. We conclude that acceptable means have been established to address potential errors in the level measurement systems under post-accident conditions.

(e) The applicant provided for audit a PG&E document 686385-3 describing type test of a valve operator with limitorque type SMB actuator and reliance motor using RH insulation. The valves tested were heat aged for 100 hours at 180°C, mechanically aged by cycling the valve through 1208 cycles, seismically tested, and exposed to 204 megarads before subjecting the valve to a LOCA environment.

The LOCA environment consisted of caustic spray, two 30-minute cycles at 300°F and 70 psig, 4 days at 250°F and 30 psig, and 26 days at 200°F and 10 psig.

Based on our review of the PG&E report, we conclude that the testing, conducted adequately envelopes the design conditions defined by the applicant for a LOCA, satisfies the requirements of IEEE 323-1971 and is therefore acceptable as a basis for environmentally qualifying these valves.

The Limitorque valve motor operators installed in Diablo Canyon Units 1 and 2 are identical to the equipment used in Salem Unit 2. Since this equipment is qualified for Salem Unit 2 containment conditions which are more severe than Diablo Canyon (T_{MAX} = 354°F versus T_{MAX} = 344°F), the application of this analysis to qualify this equipment for Diablo Canyon is appropriate. On this basis, the Diablo Canyon Limitorque valve operators for the steam line break environment satisfies the requirements of IEEE 323-1971 and is therefore acceptable.

(f) The applicant in Section 3.11.3-6 of the FSAR indicated that electrical penetrations have been successfully type tested. The type tests included temperature, pressure, and relative humidity. It appeared from this information that exposure to chemical spray, radiation preconditioning, and energizing of the penetration during simulated LOCA testing may not have been included as part of the qualification type tests as required by IEEE Standard 317-1971.

Subsequently, the applicant provided in submittal dated October 11, 1978, for audit review an additional test report number 74-502-3, "100 Series Electrical Containment Penetrations Low Voltage Qualification Test Report Addendum 1," dated March 1974.

This report indicated that the low voltage power and control, instrument signal, and thermocouple penetrations were: (1) thermally cycled 40 times from 50 to 150 to 50°F at 95 percent relative humidity at the rate of 3 to 4 cycles per day; and (2) exposed to 5×10^7 Rads at 3×10^6 r/hr before the penetrations were subjected to LOCA environment. During the subsequent LOCA test, the penetration was exposed to a steam environment with pH above 8.2 and 100 percent relative humidity for 193 hours (3 hours at 340°F and 102 psig, 3 hours at 320°F and 81 psig, 18 hours at 260°F and 25 psig, and for 169 hours at 250°F and 16 psig). The cable passing through the penetration was energized during this test.

Based on our review of the test report number 74-502-3, we conclude that the test conditions adequately enveloped the design conditions defined by the applicant for a LOCA and steam line break accident, that the tests satisfied the qualification test requirements of IEEE Standard 317-1971, and that the GE Series 100 low voltage penetrations have therefore been acceptably qualified environmentally.

With regard to GE Series 100 medium voltage (above 1000 volts) penetrations, the applicant provided four additional test reports addressing high voltage penetrations. Based on the information provided in these reports, the high voltage penetrations were exposed to 5×10^7 Rads at 3×10^6 r/hr before the penetration was subjected to a LOCA environment. During the LOCA test, the penetration was exposed to a steam environment with pH above 8.0 and 100 percent relative humidity for 4 hours at 340°F and 103 psig, 6 hours at 320°F and 81 psig, and 10 days at 260°F and 20 psig.

The high voltage penetration was not subjected (prior to LOCA tests) to thermal cycling to demonstrate the effects of I^2R heating and thermally induced stresses due to operating cycles. However, low voltage penetrations were thermally cycled prior to LOCA tests and found to be acceptable. Therefore, based on the tests performed on the low voltage penetrations and based on the epoxy being identical between the high and low voltage penetrations, we consider this item to be acceptably resolved for the GE Series 100 medium (above 1000 volts) penetrations.

The cables passing through the penetrations were not energized while the penetration was exposed to the LOCA environment. The combined effects of I^2R heating from continuous current and heating from the LOCA environment was therefore not demonstrated by these tests. Subsequently, in a letter dated May 17, 1979, the applicant provided additional information on containment electrical penetrations. This information states that 6.22°C (11.2°F) would be the temperature rise for the maximum current allowed by overcurrent protection. The 11.2°F temperature rise added to the 260°F maximum LOCA temperature environment does not exceed the 340°F temperature to which the containment penetration was tested. Therefore, we conclude that the high voltage penetration has been acceptably qualified with respect to I^2R heating for the LOCA environment. (The overcurrent protection for these penetrations has been evaluated and found acceptable as discussed later in this report.)

Qualification information provided was for the GE Series 100 penetrations. The majority of penetrations installed at Diablo Canyon are, however, GE type NS02, NS03, and NS04 penetrations. The applicant documented in a letter dated May 17, 1979, that the epoxy, in the GE type NS02, NS03, NS04, and 100 series penetrations, is chemically identical. Based on the similarity of penetrations designs, we find that the tests on the GE Series 100 penetrations are applicable to the GE type NS02, NS03, and NS04 series penetrations for environmental qualification.

(g) In Section 3.11.3-4 of the FSAR, the applicant indicated that low voltage power and control cable of the type located inside containment have undergone type tests. These tests included radiation, pressure, and temperature to simulate conditions inside

containment during a LOCA. It appeared from this information in the FSAR that exposure to chemical spray and high humidity may not have been included as part of the qualification of low voltage power and control cables.

Also in Section 3.11.3-4 of the FSAR, the applicant indicated that cable for the fan cooler motors had undergone type tests. These tests included high humidity, pressure, and temperature. Radiation tests were not performed on this cable since the cable had the same Hypalon jacket as that used for the low voltage power and control cable and since the insulation is Kapton with a gamma radiation resistance of 10^9R. It appeared from this information in the FSAR that exposure to chemical spray (in addition to radiation exposure) had not been included as part of the qualification of fan cooler cables.

In addition to the two types of cable discussed above, the applicant indicated in the FSAR that vital instrumentation and thermocouple extension wire was supplied with three different types of insulation by three different manufacturers. Also the applicant indicated that two additional types of cable supplied by different manufacturers are located outside containment and are subject to high energy line break environments. Information was not provided in the FSAR to permit an independent evaluation of the qualification of these different cable insulation systems.

The results of an audit meeting that addressed each of the above types of cable are described below.

(1) The applicant provided for audit a Continental test report (CC-21935 dated March 1971) describing type testing of cable manufactured by Continental with silicone rubber insulation. The cable tested was exposed to 281°F and 50 psi steam for 120 hours, 1×10^7 rads, 281°F and 50 psi steam chemical spray for 120 hours, and 1×10^7 rads.

    Based on our review of the Continental report, we conclude that the testing conducted adequately envelopes the design conditions defined by the applicant for a LOCA, satisfies the requirements of IEEE Standard 323-1971 and is therefore acceptable. For the steam line break environment, the applicant is presently performing a heat transfer analysis to justify that this cable can function at containment temperatures of 344°F. Pending completion of this analysis, we find the Continental cable acceptable for low power operation. For full power operation we require a satisfactory completion of the heat transfer analysis or satisfactory retesting of the cable.

(2) The applicant provided for audit a Boston Insulation Wire and Cable Company Report 9273 describing type test of cable with Ethylene propylene insulation covered with Hypalon.

    The cable tested was exposed to 1.8×10^8 rads, chemical spray, 340°F and 105 psig for one hour, 307°F and 62 psig for 5 hours, 260°F and 25 psig for 2 days, and 220°F and 7 psig for 12 days.

Based on our review of the Boston Report, we conclude that the testing conducted adequately envelopes the design conditions defined by the applicant for a LOCA and a steam line break accident, and satisfies the requirements of IEEE Standard 323-1971 and is therefore acceptable as a basis for environmental qualification of this type of cable.

(3) The applicant provided for audit a Franklin Institute Laboratory Report (F-C4033-2 dated January 1975) describing type testing of cable manufactured by Raychem with Stilan insulation.

The cable tested was aged for 7 days at 518°F and 7 days at combined 5×10^7 rads and 302°F before being subjected to a LOCA environment. During the LOCA environment, the cable was exposed to 351°F and 70 psig steam for 10 hours, 275°F and 31 psig steam for 4.5 days, 212°F and 10 psig steam/air for 26 days, 1.5×10^8 rads, and a chemical spray.

Based on our review of the Franklin Report, we conclude that the testing conducted adequately envelopes the design conditions defined by the applicant for a LOCA and a steam line break accident, satisfies the requirements of IEEE Standard 323-1971 and is therefore acceptable as a basis for environmental qualification of this type of cable.

(4) The applicant provided for audit a PG&E document 663359-69, "Tests of Electric Cables Insulated and Jackets With Tefzel 280 Fluorpolymer Under IEEE Standard 383-1974," dated August 1, 1974.

The cable tested was aged for 7 days at 356°F and 200 megarads before subjecting the cable to a LOCA environment. During the LOCA-simulated environment, the cable was exposed to the steam/pressure cycle of Table A2 of IEEE Standard 323-1974 and a chemical spray while the cable was energized.

Based on our review of the PG&E document, we conclude that the testing conducted adequately envelopes the design conditions defined by the applicant for a LOCA and a steam line break accident, satisfies the requirements of IEEE Standard 323-1971 and is therefore acceptable as a basis for environmental qualification of this type of cable.

(5) The applicant provided for audit a PG&E report LSS-1586 dated March 5. 1971, describing type tests of cable manufactured by Boston Insulation Wire and Cable Company With Silicon Glass Braid/Dapton/Hypalon insulation.

The cable tested was exposed to steam pressure of 50 $\pm$ 2 psig for 2 hours and 48 minutes, 20 $\pm$ 2 psig for 21 hours and 12 minutes, and 5 + 1 psig for 96 hours.

The tests did not include radiation preconditioning or chemical spray during testing and therefore did not meet the requirements of IEEE Standard 323-1971.

Subsequently, the applicant documented in a letter, dated November 27, 1978, additional evidence concerning environmental qualifications of the subject Boston cable. This additional evidence was reviewed and found to duplicate the information already reviewed in the FSAR and in the PG&E report provided during the July 31, 1978 through August 1, 1978 audit. The applicant was informed that the additional evidence was unacceptable and that retesting or replacement with qualified cable would be required.

The applicant provided more explicit justification in a submittal dated April 11, 1979. The justification included additional type test information. The type tests included radiation preconditioning and chemical spray tests in sequence with LOCA environment tests on separate smaller sized cable samples that used the same Kapton and Hypalon insulation materials that are used in the Boston cable. Based on the applicant's commitment to perform confirmation tests by first refueling and our review of the PG&E original report and the additional information, we conclude that there is reasonable assurance that the subject Boston cable will remain operable during and subsequent to a LOCA or steam line break accident and therefore find this type of cable acceptable for plant use during the first cycle of plant operation.

(6) The applicant provided for audit a Raychem test report EM #1030, dated September 24, 1974, describing type test of cable manufactured by Raychem with Flamtrol insulation.

The cable tested was exposed to 539°F for 1-1/2 hours while electrically energized, exposed to 539°F for 2-1/2 hours while electrically energized, passed a voltage withstand test of 4 kV, a-c, exposed to 539°F for 72 hours, electrically energized for one hour without failure, and passed a voltage withstand test of 5 kV a-c for 5 minutes.

Based on our review of the Raychem report, we conclude that the testing conducted adequately envelopes the design conditions defined by the applicant for a High Energy Line Break environment outside of containment, satisfies the requirements of IEEE Standard 323-1971 and is therefore acceptable as a basis for environmental qualifications of this type called for use outside containment.

(7) The applicant provided for audit a letter to PG&E from Okonite, dated October 14, 1974, describing type tests of cable manufactured by the Okonite company with Okonite/Okolon (Hypalon) insulation.

The cable tested was exposed to 540°F for 1.5 hours while electrically energized.

Based on our review of the Okonite letter, we conclude that the testing conducted adequately envelopes the design conditions defined by the applicant for a High Energy Line Break environment outside containment, satisfies the requirements of IEEE Standard 323-1971 and is therefore acceptable as a basis for environmental qualification of this type cable for use outside containment.

(h) In Section 3.11.3-5 of the FSAR, the applicant has indicated that all splices and terminal connections are low voltage and were made using a polyolefin heat shrinkable material that has been type tested. The type tests were performed on an assembled low voltage splice maintained at maximum rated voltage. The type tests included (a) heat aging at 121°C for 168 hours, (b) both 100 and 200 Mrads, (c) 5 hours at 360°F and 70 psig steam, (d) 6 hours at 320°F and 70 psig steam, (e) 24 hours at 250°F, 21 psig steam, and chemical spray, and (f) 12 days at 221°F and 2.5 psig steam.

These parameters have been described in the FSAR and are representative of the service conditions to which electrical connections located inside containment may be exposed to during and following a design basis accident. However, acceptance criteria, test results, and the basis for acceptability were not presented in the FSAR to permit an independent evaluation of the qualification as required by Section 4.4 of IEEE Standard 323-1971. We therefore required that the qualification program information be provided at the audit meeting to show that the Diablo Canyon cable connections and terminations are adequately qualified for this intended application.

During the audit meeting, the applicant provided PG&E test report DC 663359-16-1 describing the above type tests. Since acceptance criteria, test results and basis for acceptability were not explicitly presented in the test report, the applicant provided an additional test report (Franklin test report number F-C4033-3 dated January 1975), describing type tests on Electrical Connections.

According to this Franklin test report, the electrical connections tested were aged for 7 days in a combined 302°F and 5×10^7 rads before subjecting the connection to a LOCA environment. During the LOCA-simulated environment, the connection was exposed to 351°F and 70 psig steam for 10 hours, 275°F at steam pressure for 4.5 days, 212°F at steam/air pressure of 10 psig for 26 days, an addition 1.5×10^8 rads, and chemical spray.

Based on our review of the Franklin report, we conclude that the testing conducted adequately envelopes the design conditions defined by the applicant for a LOCA and steam line break accident, satisfies the requirements of IEEE Standard 323-1971, and is therefore acceptable as a basis for environmental qualification of this type of electrical connectors and splices when encased in this polyolefin heat shrinkable material.

(i) The safety function of stem-mounted limit switches is to provide the inputs for valve position indication in the main control room. The applicant in a letter dated May 3, 1978, indicated that such stem mounted limit switches are located inside containment, are subject to LOCA environment, and have been shown to be qualified. We therefore required that the qualification program information be provided during the audit meeting to show that these stem-mounted limit switches are adequately qualified for their intended applications.

The applicant provided for audit a NAMCo test report, "Qualification of NAMCo Limit Switches Model EA-180 to IEEE Standards 344(75), 323(74), and 382(72)," dated March 3, 1978, describing type test of a limit switch manufactured by NAMCo.

The limit switch tested was aged for 200 hours at 200°F, 100,000 actuations under electrical load, 204 megarads, and seismically shaken before subjecting the switch to a LOCA environment. During the LOCA-simulated environment, the switch was exposed to 340°F and 70 psi for 3 hours, 320°F and 40 psi for 2 hours, 250°F and 25 psi for 3.5 days, 200°F and 10 psi for 26 days, and chemical spray.

Based on our review of the NAMCo report, we conclude that testing conducted adequately envelopes the design conditions defined by the applicant for a LOCA and steam line break accident, satisfies the requirements of IEEE Standard 323-1971, and is therefore acceptable as a basis for environmental qualification of these switches.

(j) Auxiliary feedwater isolation valves were identified as being required to operate in a steam line break environment outside containment. However, the information provided in the FSAR did not delineate specific reference to their qualification documentation. Therefore, we requested, and the applicant provided, a Limitorque environmental qualification report 8003 for Limitorque Valve Actuator (type SMB) equipped with an electric motor with Class B insulation.

The valve tested was heat aged for 200 hours at 165°F and 100 percent relative humidity, mechanically aged by cycling the valve through 2000 cycles, exposed to 2.04×10^8 rads, and seismically tested before subjecting the valve to a steam line break environment.

The steam line break qualification environment consisted of one 10-second temperature rise transient from 120°F with the temperature held at 250°F for 30 minutes, one 10-second temperature rise transient from 120° to 250°F with the temperature of 250°F held for 24 hours, and 16 days at 200°F.

Based on the applicant's statement that the materials of construction in the SMC-04 type valve are the same as those in the SMB type valve (documented in PG&E letter dated June 17, 1979) and based on our review of Limitorque report B003, we conclude that the testing conducted adequately envelopes design conditions for the steam line break outside containment and satisfies the requirements of IEEE-323-1971 and is acceptable.

(k) The applicant identified a new item, conduit seals, that are needed to seal the interface between Rosemont pressure transmitters or limit switches and their conduit cable connections. The seal is required to validate the environmental test performed for the Rosemont pressure transmitters and limit switches.

The applicant was requested to provide the environmental qualification report for the conduit seals. Pursuant to our request, the applicant provided a Conax Corporation test report IPS-409, Qualification Report for Conductor modules for Arkansas Nuclear One, Unit 2. The seal tested was mechanically aged by cycling the seal through 5 cycles from 30°F to 150°F and 120 cycles from 35°F to 145°F, heat aged for 169 hours at 249.8°F, exposed to 2.238×10^8 rads, and seismically tested before subjecting the seal to a LOCA environment.

The qualification test environment consisted of two 3-hour cycles at 340°F, 3 hours at 320°F, 4 hours at 300°F and 30 days at 250°F. In addition, the conductors passing through the seal were energized and the seal was exposed to 24 hours of chemical spray.

Based on our review of Conax Corporation's test report IPS-409, we conclude that the testing performed adequately envelopes the design conditions for both steam line break and LOCA exceeds the qualification test requirements of IEEE-323-1971 and is therefore acceptable as a basis for environmental qualification of these seals.

Summary of Environmental Qualifications of Electrical Equipment

In addition to the above, we have recently published staff guidance to be used in environmentally qualifying electrical equipment (see NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment"). Recognizing that the equipment qualification review for the Diablo Canyon plant has been a long-term effort spanning several years, we recently required that PG&E reassess its qualification documentation for equipment installed at Diablo Canyon with the purpose of establishing that the qualification methods used and results obtained are in conformance with the staff positions contained in NUREG-0588. We believe based on our reviews conducted to date as discussed above that this additional review will confirm our earlier conclusions regarding the adequacy of the qualification documentation. On May 27, 1980, the Commission issued a Memorandum and Order which directed the NRC staff to complete its review of environmental qualification, including the publication of safety evaluation reports by February 1, 1981, and that all safety-related electrical equipment in all operating plants shall be qualified in accordance with the previous Division of Operating Reactors (DOR) Guidelines or NUREG-0588 by June 30, 1982. We will complete review of this matter for Diablo Canyon Units 1 & 2 on a schedule that conforms with the above requirements.

8.0 ELECTRIC POWER

8.1 General

Low and/or Degraded Grid Voltage Condition

As reported in SER Supplement 6, we requested the applicant to provide additional information in regard to the effects of short term or long term degradation of offsite power voltage on safety related equipment. The staff has since developed additional requirements concerning (a) sustained degraded voltage conditions at the offsite power source, and (b) interaction of the offsite and onsite emergency power systems. These additional requirements were defined in the following staff positions.

1. We require that a second level of voltage protection for the onsite power system be provided and that this second level of voltage protection shall satisfy the following requirements:

   a) The selection of voltage and time set points shall be determined from an analysis of the voltage requirements of the safety-related loads at all onsite system distribution levels;

   b) The voltage protection shall include coincidence logic to preclude spurious trips of the offsite power source;

   c) The time delay selected shall be based on the following conditions:

      (i) The allowable time delay, including margin, shall not exceed the maximum time delay that is assumed in the FSAR accident analyses;

      (ii) The time delay shall minimize the effect of short duration disturbances from reducing the availability of the offsite power source (s);

      (iii) The allowable time duration of a degraded voltage condition at all distribution system levels shall not result in failure of safety systems or components;

      (iv) The voltage sensors shall automatically initiate the disconnection of offsite power sources whenever the voltage set point and time delay limits have been exceeded;

      (v) The voltage sensors shall be designed to satisfy the applicable requirements of IEEE Std. 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations;" and

(vi) The Technical Specifications shall include limiting condition for operation, surveillance requirements, trip setpoints with minimum and maximum limits, and allowable values for the second- level voltage protection sensors and associated time delay devices.

2. We require that the system design automatically prevent load shedding of the emergency buses once the onsite sources are supplying power to all sequenced loads on the emergency buses. The design shall also include the capability of the load shedding feature to be automatically reinstated if the onsite source supply breakers are tripped. The automatic bypass and reinstatement feature shall be verified during the periodic testing identified in Position 3.

   In the event an adequate basis can be provided for retaining the load shed feature when loads are energized by the onsite power system, we will require that the setpoint value in the Technical Specifications, which is currently specified as "equal to or greater than..." be amended to specify a value having maximum and minimum limits. The applicant's bases for the setpoints and limits selected must be documented.

3. We require that the Technical Specifications include a test requirement to demonstrate the full functional operability and independence of the onsite power sources at least once per 18 months during shutdown. The Technical Specifications shall include a requirement for tests: (a) simulating loss of offsite power, (b) simulating loss of offsite power in conjunction with a safety injection actuation signal; and (c) simulating interruption and subsequent reconnection of onsite power sources to their respective buses.

4. The voltage levels at the safety-related buses should be optimized for the full load and minimum load conditions that are expected throughout the anticipated range of voltage variations of the offsite power source by appropriate adjustment of the voltage tap settings of the intervening transformers. We require that the adequacy of the design in this regard be verified by actual measurement, and by correlation of measured values with analysis results.

   The following items address the resolution of problem areas revealed during our review of the design conformance with the positions noted above.

   (1) There are three redundant and independent onsite emergency buses. Each bus has two levels of undervoltage protection.

   The first level consists of three undervoltage relays with their adjustable trip setpoints. Two of the three relays are connected in a two out of two logic arrangement so that when the trip setpoints on both relays have been exceeded, it will result in the immediate shedding of loads from the emergency bus and a delay disconnection from the onsite power supply. The third relay initiates diesel generator start when its trip setpoints are exceeded.

The second level consists of one undervoltage relay and two timers each with their adjustable trip setpoints. When the trip setpoint on the undervoltage relay has been exceeded the two timers are started. When the timers' setpoints have been exceeded the diesel generator starts, loads are immediately shed from the emergency bus, and the offsite power supply is disconnected after a delay.

We reviewed the implementation of this design, as depicted on electrical logic and schematic diagrams and conclude that the design meets our position that the design did not include coincident logic per emergency bus to preclude spurious trips of the offsite power source. We informed the applicant of our requirement and the applicant agreed to implement this feature. We have reviewed the implementation on electrical logic diagrams and conclude that with this modification the design is now acceptable with respect to this problem area.

(2) Our initial review of the electrical schematics revealed that the load shedding feature was not bypassed during sequencing of loads on the emergency bus or when power is being supplied to the emergency bus by the diesel generator as required by our position number 2. The applicant agreed to implement this feature in the design. We have reviewed the implementation on electrical logic diagrams and concluded that the design is now acceptable with respect to this problem area.

(3) Compliance with other aspects of these positions such as surveillance, Technical Specifications, test requirements, and optimization of the transformer voltage tap settings have also been documented by the applicant. Based on our review of this documentation we have determined that the design now satisfies our above positions and is acceptable with respect to low and/or degraded grid voltage condition.

Overcurrent Fault Protective Systems for Containment Penetrations

As reported in our supplemental safety evaluation report No. 8 dated October 20, 1978, Regulatory Guide 1.63, Revision 1 was classified as a Category II review item by the Regulatory Requirements Review committee meeting No. 60, March 27, 1977, for all applications not evaluated under Revision 0 to Regulatory Guide 1.63.

Pursuant with this Category II classification, the Diablo Canyon applicant was requested to describe how their penetration design meets Regulatory Guide 1.63, Revision 1.

In this regard the applicant was requested to:

a. Identify each type of electrical circuit that penetrates containment.

b. Describe the primary and backup over-current protective systems provided for each type of circuit identified in Item 1(a).

c. Describe the fault-current - versus-time for which the primary and backup over current protective systems are designed and qualified.

d. Describe the current - versus-time for which the penetrations are designed and qualified.

e. Provide coordinated curves between Items 3(c) and 3(d) for each circuit identified in Item 3(a) to show that the fault-current - versus-time condition to which the penetrations is qualified will not be exceeded.

f. Describe the provisions for periodic testing under simulated fault conditions.

In order to demonstrate compliance with the position outlined above, the applicant provided additional information on fault current protection for containment electrical penetrations by letters dated February 26, 1979 and September 5, 1979. Based on our review of this additional information, we conclude that the information adequately addresses overcurrent protection as it relates to the qualifications of containment penetrations and that the Diablo Canyon design is in compliance with the position outlined above and, therefore, this overcurrent protection is acceptable.

9.0 AUXILIARY SYSTEMS

9.6.1 Fire Protection

In SER Supplement 8 we stated the following items had not been resolved:

(1) The applicant has not electrically supervised all valves in the fire water system that are necessary to ensure the fire water supply to the areas containing safety related equipment.

(2) The applicant has not shown that alternate plant shutdown capability is independent of the cable spreading room and control room.

(3) The applicant has not demonstrated that a failure of recently installed non-seismic fire protection equipment will not affect the safe cold shutdown of the plant.

(4) The applicant has installed some fire barrier cable penetration seals that have not passed a three-hour fire barrier test.

(5) The applicant's small scale fire tests of Pyrocrete 102 are not representative of actual field installation.

Our evaluation and conclusions are noted below for each of the items listed:

(1) The Branch Technical Position ASB BTP 9.5.1 requires that key valves in the fire water supply system be electrically supervised or that they be locked open. In submittal dated July 18, 1979, the applicant has committed to lock open the above cited valves. We find that this alternate means of assuring the supply of fire water to areas containing safety related equipment acceptable. We consider this matter resolved.

(2) In a submittal dated November 13, 1978 the applicant provided information that demonstrated that the alternate shutdown capability is independent of the cable spreading and control room and would not be affected by either a fire in the control room or in the cable spreading room. Based on our review of the above information, we find that in the event of a fire in the control room or the cable spreading room the plant can be brought to a cold shutdown using the hot shutdown panel.

(3) In the submittal dated November 13, 1978, the applicant committed to seismically supporting non-seismic fire protection equipment in safety related areas so that this equipment will remain in place after the safe shutdown earthquake. We have reviewed the applicant's design and find the installation of the recently installed fire protection equipment acceptable in this regard.

(4) In the submittal dated November 13, 1978 the applicant referenced acceptable standards to which the electrical penetrations were fire tested. The test conducted by the applicant demonstrated that the penetrations have a three hour rating. Therefore, the fire barriers cable penetration seals are now acceptable.

(5) The applicant described in detail the field installation of the Pyrocrete 102 fire barriers and referenced the fire test results on Pyrocrete barriers of similar construction. Based on the test results of similar barriers, we now find the Pyrocrete 102 fire barriers acceptable.

In Supplement 8 we reported that the applicant had committed to install Seismic Category I fire dampers in the supply ducting to the 4k switchgear rooms. These fire dampers have now been installed, therefore we consider this matter resolved. All matters relating to the fire protection program have now been resolved.

15.0 ACCIDENT ANALYSIS

15.2.6 Anticipated Transient Without Scram (ATWS)

We have reviewed the PG&E submittal of March 9, 1979, on Emergency Operating Procedures for the postulated anticipated transients without scram (ATWS) events. We provided our comments on the proposed procedures and made recommendations for changes. The proposed procedures must be modified in accordance with our comments and instructions to be acceptable for full power operation. However, the Diablo Canyon plant may be operated at low power (less than or equal to five percent of full power) prior to completion of procedures modifications without undue risk to the health and safety of the public. Our conclusion that low power operation is acceptable is based on our understanding of the expected plant response to the relevant ATWS events under these limited operating conditions.

17.0 QUALITY ASSURANCE

Our review of the quality assurance program description for the operations phase for the Diablo Canyon Nuclear Plant has verified that the criteria of Appendix B to 10 CFR Part 50 have been adequately addressed in Section 17.2 of the FSAR through Amendment 81. This determination of acceptability included a review of the list of safety-related structures, systems, and components (Q-list) to which the quality assurance program applies. The staff has recently developed a revised procedure for conducting the Q-list review that involves other NRR technical review branches and significantly enhances the staff's confidence in the acceptability of the Q-list. Staff re-review of the Q-list using the revised procedure is presently underway and the results will be reported in a later supplement. This re-review is not considered to be of sufficient importance to require its completion prior to granting authority to load fuel and perform low power tests.

APPENDIX A
CHRONOLOGY OF RADIOLOGICAL REVIEW

Date Docketed	Description of Documents
January 2, 1978	Letter from PG&EC transmitting additional information regarding environmental qualification of reactor coolant temporary detectors and containment pressure transmitters.
November 10, 1978	Letter from Westinghouse Electric Corporation transmitting information on seismic qualification for all Westinghouse valves.
November 13, 1978	Letter to PG&EC transmitting NRC Fire Protection Review questions 5, 8, 9, 21, 23, 24, 26-29, 45, 46, and 49-51.
November 14, 1978	Letter to PG&EC transmitting the ACRS Report dated November 7, 1978.
November 14, 1978	Letter to PG&EC transmitting Safety Evaluation Report Supplement No. 8.
November 20, 1978	Letter from PG&EC transmitting Amendments 72 and 73 to the FSAR.
November 21, 1978	Letter to PG&EC on Staff Evaluation of Probabilistic Seismic Risk Assessment.
November 27, 1978	Letter from PG&EC providing additional information on Environmental Qualification of Fan Cooler Power Cables.
November 28, 1978	Letter from PG&EC on Seismic Test Procedure for 4160V Class 1E Switchgear Bus Undervoltage Relays.
November 30, 1978	Letter from PG&EC on Seismic Test Procedure for Diesel Generator Control Cabinet.
December 1, 1978	Letter from PG&EC on Seismic Test Procedures on certain electrical equipment.
December 6, 1978	Letter from PG&EC transmitting request for exemption from ASME Code Section 11 pump and valve testing requirements.
December 6, 1978	Letter from PG&EC providing additional information on Seismic Test Procedures for electrical components.

December 6, 1978	Letter to PG&EC transmitting Amendments No. 1 and 4 to CPPR-39 and CPPR-69, Federal Register notice and safety evaluation supporting amendments.
December 7, 1978	Letter from PG&EC concerning NEMA size 2 motor controllers.
December 7, 1978	Letter from PG&EC on large-break loss-of-coolant (LOCA) accident.
December 18, 1978	Letter from PG&EC providing additional information on steam line breaks.
December 19, 1978	Letter from PG&EC forwarding analysis of large break loss-of-coolant accident.
December 19, 1978	Letter from PG&EC providing additional information on overpressure protection.
December 19, 1978	Letter from PG&EC forwarding additional seismic information in regard to certain safety-related equipment.
December 21, 1978	Letter to PG&EC in regard to the Depth/Thickness Ratio Deficiency for Turbine Building Column Webs.
December 22, 1978	Letter from PG&EC forwarding letter from fire protection consultants Gage-Babcock.
December 27, 1978	Letter from PG&EC forwarding responses to questions 220.1 thru 220.6 of staff letter of July 25, 1978.
January 3, 1979	Letter from PG&EC providing additional information on seismic qualification of electrical equipment.
January 4, 1979	Letter from PG&EC providing additional information on seismic retesting of class IE electrical equipment.
January 4, 1979	Letter from PG&EC transmitting seismic test procedure for fan cooler motor controller.
January 12, 1979	Letter from PG&EC fowarding additional information regarding containment polar cranes and turbine building cranes.
January 12, 1979	Letter from PG&EC forwarding additional information in regard to water hammer test.

February 2, 1979	Letter from PG&EC forwarding description of fireproofing for electrical conduits.
February 5, 1979	Letter from PG&EC transmitting Amendments 75 and 76 to the FSAR.
February 5, 1979	Letter to PG&EC in regard to mutually redundant class I circuits located in the control room panels.
February 26, 1979	Letter from PG&EC forwarding additional material requesting fault current protection for containment electrical penetrations.
March 1, 1979	Letter from PG&EC forwarding information regarding tests of modeled circuit breakers, ventilation control logic cabinet and fan cooler controllers.
March 7, 1979	Letter from PG&EC providing information on natural circulation boron mixing test procedure, confirmation of fire water tank piping design criteria, and fault current protection for
March 9, 1979	Letter from PG&EC forwarding emergency operating ATWS procedures.
March 9, 1979	Letter from PG&EC forwarding addendum 2 to Wyle Laboratory Test report on seismic testing on safety-related electrical equipment.
March 14, 1979	Letter to PG&EC transmitting request for additional information regarding Main Steam Line Break Inside Containment.
March 14, 1979	Letter to PG&EC transmitting request for additional information regarding containment purging isolation valves.
May 16, 1979	Letter to PG&EC transmitting request for additional information regarding environmental qualification matters.
March 19, 1979	Letter to PG&EC transmitting IE Bulletin 78-128.
March 28, 1979	Letter from PG&EC transmitting test report on Barton Transmitters.
March 29, 1979	Letter from PG&EC transmitting Amendments 77 and 78.
April 12, 1979	Letter from PG&EC transmitting information on environmental qualification of fan cooler power cables.
April 13, 1979	Letter from PG&EC providing information on containment purge valves.

April 13, 1979	Letter to PG&EC transmitting Order Extending Construction Permit Completion Dates.
May 17, 1979	Letter from PG&EC forwarding additional information on containment electrical penetrations.
May 25, 1979	Letter to PG&EC transmitting request for additional information regarding instrumentation qualification.
June 11, 1979	Letter from PG&EC transmitting information on environmental qualification on certain electromedical devices.
June 14, 1979	Letter from PG&EC submitting additional information on environmental qualification of Barton pressure and differential pressure transmitters.
June 14, 1979	Letter from PG&EC transmitting additional information on steam line break inside containment.
June 15, 1979	Letter from PG&EC transmitting additional information on Westinghouse ECCS Model.
July 2, 1979	Letter from PG&EC transmitting information in high voltage penetration thermal cycling, fault current protection for containment penetrations, fire protection programs and inservice testing.
July 2, 1979	Letter to PG&EC transmitting request for additional information regarding refueling water storage tank (RWST) capacity.
July 18, 1979	Letter from PG&EC submitting information on the fire protection program.
August 2, 1979	Letter to PG&EC requesting additional information on secondary water chemistry.
August 2, 1979	Letter to PG&EC transmiting request for additional information regarding certain Westinghouse Corporation reports.
August 3, 1979	Letter to PG&EC requesting additional information on loss-of-coolant accident.
August 20, 1979	Letter from PG&EC transmitting Amendments 79 and 80 to FSAR and Hosgri Report.
August 27, 1979	Letter to PG&EC requesting additional information on small break accidents.

September 5, 1979	Letter from PG&EC providing information on primary overcurrent fault protection.
September 7, 1979	Letter from PG&EC on environmental qualification of Barton transmitters.
September 12, 1979	Letter to PG&EC requesting additional information in regard to structural acceptability of certain components.
September 27, 1979	Letter from PG&EC requesting exemption from Appendix J leakage testing program requirements.
September 27, 1979	Partial Initial Decision, Operating Licensing Proceedings.
October 5, 1979	Letter to PG&EC transmitting request for additional information regarding potential overpressurization of containment in the event of main steam line break.
October 5, 1979	Letter to PG&EC transmitting request for additional information.
October 17, 1979	Letter to PG&EC transmitting information requesting anticipated transients without scram for light water reactors.
October 17, 1979	Letter from PG&EC transmitting information regarding fuel handling building crane, polar crane, turbine building and intake structure crane.
November 2, 1979	Letter to PG&EC transmitting request for additional information on Environmental Qualification of Class IE Instrumentation and Electrical Equipments.
November 7, 1979	Letter from PG&EC transmitting information on relief valve actuator.
November 8, 1979	Letter from PG&EC transmitting justification for temperature of continental cable.
November 8, 1979	Letter to PG&EC transmitting information regarding Environmental Qualification of Reactor Coolant Temperature Detectors and Containment Pressure Transmitters.
November 30, 1979	Letter to PG&EC regarding separation of electrical equipment.
December 5, 1979	Letter from PG&EC providing information on secondary water chemistry control.

December 5, 1979	Letter from PG&EC transmitting schedule for providing information regarding environmental qualification of Class IE Instrumentation and electrical equipment.
December 5, 1979	Letter from PG&EC providing information on secondary water chemistry control.
December 18, 1979	Letter from PG&EC submitting Amendments No. 81 and 82.
January 25, 1979	Letter from PG&EC submitting Amendment No. 74 to FSAR updated.
December 28, 1979	Letter from PG&EC providing results of evaluation to detect effects of moderate energy line breaks on equipment required for safe shutdown.
February 29, 1980	Letter to PG&EC providing information regarding field audit of FSAR separation criteria for electric equipment.
January 14, 1980	Letter from PG&EC in regard to qualification of Class IE electrical equipment.
January 22, 1980	Letter from PG&EC forwards information regarding temporary effects on steam generator level measurements.
February 5, 1980	Letter to PG&EC regarding issuance of NUREG-0588, Interim Staff Position on Equipment Qualification of Safety-Related Electrical Equipment.
February 13, 1980	Letter to PG&EC regarding rod drop protection.
February 21, 1980	Letter to PG&EC requesting additional information on safety-related electrical equipment.
March 3, 1980	Letter to PG&EC regarding change in review procedures for equipment qualification documentation for the Diablo Canyon Nuclear Power Plants 1 and 2.
March 17, 1980	Letter from PG&EC providing information on rod drop protection.
March 20, 1980	Letter from PG&EC forwarding additional information regarding refueling water storage tank capacity.

March 25, 1980	Letter from PG&EC fowarding proprietary version of response to NRC September 12, 1979 letter regarding tube wear.
April 8, 1980	Letter from PG&EC providing information on secondary water chemistry.
April 16, 1980	Letter from PG&EC providing additional information regarding refueling water storage tank capacity.
May 7, 1980	Letter from PG&E providing information on systems interactions study.
May 8, 1980	Letter to PG&E in regard to plugging of row 1 steam generator tubing and installation of steam generator inspection pools.
May 27, 1980	Letter from PG&E providing information on systems interactions study.

APPENDIX B
NUCLEAR REGULATORY COMMISSION
UNRESOLVED SAFETY ISSUES

The Nuclear Regulatory Commission's Atomic Safety and Licensing Appeal Boards (ASLAB) have issued two decisions that address the NRC staff's responsibilities for considering unresolved generic safety questions in the context of specifc licensing proceedings. These decisions were ALAB-444 issued on November 23, 1977, in the matter of the Gulf States Utility Company application for construction permits for the River Bend Station, Unit Nos. 1 and 2, and ALAB-491 issued on August 25, 1978, in the matter of Virginia Electric Power Company's application for operating licenses for the North Anna Power Stations, Units 1 and 2.

In accordance with a memorandum by the Atomic Safety and Licensing Board for Diablo Canyon dated September 18, 1978, an affidavit[1] was filed by the NRC staff that responded to the requirements of these two ASLAB decisions.

The affidavit of Messrs. Aycock, Crocker, and Allison addressed 23 generic safety issues as they related to the operation of the Diablo Canyon units. The issues addressed were those Priority Category A safety issues in the NRC Generic Issues Program that were applicable and relevant to the Diablo Canyon Units. The generic tasks addressing these safety issues are listed in Table B-1.

Since February 1979, the NRC staff has limited its discussion of generic safety issues in Safety Evaluation Reports to those issues that have been designated by the NRC staff and the Commissioners as "Unresolved Safety Issues" (USI) for reporting to the Congress pursuant to Section 210 of the Energy Reorganization Act of 1974, as amended. The USIs relevant to operation of the Diablo Canyon units are noted by asterisks in Table B-1.

Since each of these USIs was addressed previously on the Diablo Canyon hearing record in the affidavit of Messrs. Aycock, Crocker, and Allison, we have not attempted to recapitulate that information in this appendix. We have reviewed the affidavit, however, to determine if additional pertinent information has come to light since that time. We have judged that additional information should be provided to update the affidavit with respect to four of

[1] Affidavit of Michael B. Aycock, Lawrence P. Crocker, and Dennis P. Allison Relating to the Status of NRC Staff Activities Regarding Generic Safety Issues in the Matter of Pacific Gas and Electric Company (Diablo Canyon Nuclear Generating Station, Units 1 and 2) dated February 5, 1979.

TABLE B-1

List of Generic Tasks
Addressed in February 1979 Affidavit

Task No	Title
*A-1	Wate Hammer
*A-2	Asymmetric Blowdown Loads on BWR Primary Coolant Systems
*A-3	Westinghouse Steam Generator Tube Integrity
*A-9	Anticipated Transients Without Scram2
*A-11	Reactor Vessel Materials Toughness
*A-12	Fracture Toughness of Steam Gènerator and Reactor Coolant Pump Supports
A-13	Snubber Operability Assurance
A-14	Flaw Detection
*A-17	Systems Interaction in Nuclear Power Plants
A-18	Pipe Rupture Design Criteria
A-21	Main Steam Line Break Inside Containment - Evaluation of Environmental Conditions for Equipment Qualification
A-22	PWR Main Steam Line Break - Core, Reactor Vessel and Containment Response
A-23	Containment Leak Testing
*A-24	Qualification of Class IE Safety-Related Equipment
*A-26	Reactor Vessel Pressure Transient Protection
A-29	Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage
A-30	Adequacy of Safety-Related DC Power Supplies
*A-31	RHR Shutdown Requirements
A-32	Missile Effects
A-35	Adequacy of Offsite Power Systems
*A-36	Control of Heavy Loads Near Spent Fuel
*A-43	Containment Emergency Sump Reliability
*A-44	Station Blackout

2The issue of Anticipated Transients Without Scram was not specifically addressed in the affidavit of Aycock, Crocker, Allison of February 15, 1979. It was addressed in the testimony of Ashok Thadani dated March 2, 1979.

these generic tasks. The sections in this SER supplement where the additional information on three of these tasks can be found are referenced in Table B-2. The additional information on the fourth task is provided below.

Generic Task A-44 Station Blackout

Electrical power for safety systems at nuclear power plants must be supplied by at least two redundant and independent divisions. The systems used to remove decay heat to cool the reactor core following a reactor shutdown are included among the safety systems that must meet these requirements. Each electrical division for safety systems includes an offsite alternating current (ac) power connection, a standby emergency diesel generator ac power supply, and direct current (dc) sources.

Task A-44 involves a study of whether or not nuclear power plants should be designed to accommodate a complete loss of all ac power, i.e., a loss of both the offsite and the emergency diesel generator ac power supplies. A loss of all ac for an extended period of time in pressurized water reactors accompanied by loss of the auxiliary feedwater pumps (usually one of two redundant pumps is a steam turbine driven pump that is not dependent on ac power for actuation or operation) could result in an inability to cool the reactor core, with potentially serious consequences. This particular accident sequence was a significant contributor to the overall risk associated with the PWR analyzed in the Reactor Safety Study (WASH-1400). The steam turbine driven auxiliary feedwater pump for the PWR analyzed in WASH-1400 had no ac dependencies. If the auxiliary feedwater pumps are dependent on ac power to function, then a loss of all ac power could of itself result in an inability to cool the reactor core and, accordingly, this event sequence would be expected to be more important to the overall risk posed by the facility.

A loss of all ac power was not a design basis event for Diablo Canyon Units 1 and 2. Nonetheless, the combination of design, operation, and testing requirements that have been imposed on the applicant will assure that these units will have substantial resistance to a loss of all ac and that even if a loss of all ac should occur, there is reasonable assurance that the core will be cooled. These are discussed below.

A loss of offsite ac power involves a loss of both preferred offsite power sources. Our review and basis for acceptance of the design, inspection, and testing provisions for the offsite power system are described in Section 8.2 of the Diablo Canyon Units 1 and 2 SER. In addition, the applicant conducted a grid stability analysis. Our review of this analysis is described in Section 8.2 of the SER.

If offsite power is lost, the design of the onsite power system is arranged such that five diesel generators are used to provide onsite power to six Class IE distribution systems (three independent and redundant distribution systems per unit).

Our review of the design, testing, surveillance, and maintenance provisions for the Diablo Canyon Units 1 and 2 onsite emergency diesels are described in Section 8.3.1 of the SER and

TABLE B-2

Affected Generic Tasks and Sections of Accompanying SER Supplement Where Updated Discussions of "Unresolved Safety Issues" Are Provided

Task No.	Title	SER Supplement Sections
A-9	ATWS	15.0
A-11	Reactor Vessel Material Toughness	5.2
A-24	Qualificaton of Class IE Safety-Related Equipment	7.8

Section 8.3 of SER Supplement 7. Our requirements include preoperational and periodic testing to assure the reliability of the installed diesel generators in accordance with the provisions of Regulatory Guide 1.108. In addition, the applicant has been requested to implement a program for enhancement of diesel generator reliability to better assure the long-term reliability of the diesel generators.

Even if both offsite and onsite ac power sources are lost, cooling water can still be provided to the steam generators by the auxiliary feedwater system by employing a steam turbine driven pump that does not rely on ac power for operation. Our review of the auxiliary feedwater system design and operation is described in Section 10.5 of the Diablo Canyon Units 1 and 2 SER and Section 10.5 of Supplements 7 and 8. Additional actions by the NRC staff and the applicant to improve the reliability of the auxiliary feedwater systems for Diablo Canyon Units 1 and 2 are described in Section II.K.1 of the separate SER Supplement covering TMI-2 related requirements for near-term operating license applications.

In addition, we are requiring the applicant to perform analyses of accidents and transients and to develop operating guidelines, operating procedures, and conduct operator training based on these analyses as described in this supplement in Section I.C.1 of the separate SER supplement dealing with TMI-2 related matters. These requirements will include consideration of loss of all ac power in its low power test program as described in Section I.G of the latter report.

Based on the foregoing, we have concluded that there is reasonable assurance that Diablo Canyon Units 1 and 2 can be operated prior to the ultimate resolution of this generic issue without endangering the health and safety of the public.

Based on the information presented in the February 5, 1979 affadavit of Messrs. Aycock, Crocker, and Allison and the additional information provided above and in the accompanying SER supplement (as referred to in Table B-2), we have concluded that Diablo Canyon Units 1 and 2 can be operated prior to ultimate resolution of all of the asterisked generic issues in Table B-1 without undue risk to the health and safety of the public.

NRC FORM 335 (7-77)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by DDC): NUREG-0675

4. TITLE AND SUBTITLE (Add Volume No., if appropriate): Safety Evaluation Report related to operation of Diablo Canyon Nuclear Power Station, Units 1 and 2, Pacific Gas and Electric Company, Supplement No. 9

2. (Leave blank)

3. RECIPIENT'S ACCESSION NO.

7. AUTHOR(S)

5. DATE REPORT COMPLETED: MONTH June, YEAR 1980

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code):
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

DATE REPORT ISSUED: MONTH June, YEAR 1980

6. (Leave blank)

8. (Leave blank)

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code): Same as 9. above

10. PROJECT/TASK/WORK UNIT NO.

11. CONTRACT NO.

13. TYPE OF REPORT

PERIOD COVERED (Inclusive dates)

15. SUPPLEMENTARY NOTES: Docket Nos. 50-275 and 50-323

14. (Leave blank)

16. ABSTRACT (200 words or less)

Supplement No. 9 to the Safety Evaluation Report for Pacific Gas and Electric Company's application for licenses to operate the Diablo Canyon Nuclear Power Station (Docket Nos. 50-275 and 50-323) located in San Luis Obispo County, California has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. The purpose of this Supplement is to update the Safety Evaluation Report by providing the staff's evaluation of additional information submitted by the applicant since the issuance of Supplement No. 8 to the Safety Evaluation Report. Subject to favorable resolution of the items discussed in the Safety Evaluation Report and its supplements, the staff concludes that the plant can be operated by the Pacific Gas and Electric Company without endangering the health and safety of the public.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT

19. SECURITY CLASS (This report)

20. SECURITY CLASS (This page)

21. NO. OF PAGES

22. PRICE $

NRC FORM 335 (7-77)

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

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

POSTAGE AND FEES PAID
U.S. NUCLEAR REGULATORY
COMMISSION

U.S. MAIL

NUREG-0675

SER RELATED TO THE OPERATION OF DIABLO CANYON NUCLEAR POWER STATION, UNITS 1 & 2

JUNE 1980