November 18, 1969

SAFETY EVALUATION

BY THE

DIVISION OF REACTOR LICENSING

UNITED STATES ATOMIC ENERGY COMMISSION

IN THE MATTER OF

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR POWER PLANT

UNIT 2

DOCKET NO. 50-323

TABLE OF CONTENTS

TABLE OF CONTENTS (cont'd)

1.0 INTRODUCTION

The Pacific Gas and Electric Company (PG&E), by application dated July 1, 1968, and subsequent amendments, requested a license to construct and operate a second pressurized water reactor at its Diablo Canyon site which is located in San Luis Obispo County, California.

The proposed reactor is designed to operate at 3250 MW(t) with an expected ultimate capability of producing 3580 MW(t). The applicant has designed the major components including the containment structure and emergency core cooling system for a power level of 3580 MW(t), and has used this power level in analyzing postulated accidents in conformance to the guidelines of 10 CFR Part 100. We have evaluated the containment and emergency cooling systems for 3580 MW(t); however, the thermal and hydraulic characteristics of the reactor core were evaluated at 3250 MW(t). Before operation at any power level above 3250 MW(t) is authorized by the Commission, the Commission must perform a safety evaluation to assure that the facility can be operated safely at the higher power level.

Several facilities and items of equipment will be shared between Diablo Canyon Unit 1, for which a permit to construct has already been issued, and the proposed Unit 2. Our review of these shared facilities and items indicates that the independence of either Unit with regard to safety would not be compromised by a failure of the shared items. All shared items important to public health and safety are suitably backed-up by redundant equipment.

Our technical safety review of the proposed plant has been based on the applicant's Preliminary Safety Analysis Report (PSAR) and the six subsequent amendments, all of which are contained in the application. The

technical evaluation of the preliminary design of the proposed plant was accomplished by the Division of Reactor Licensing with assistance from the various consultants, as requested. Within Reactor Licensing, the Reactor Projects group was responsible for the review, for coordinating parts of the review involving personnel with various special technical disciplines from the Reactor Technology and Reactor Operations groups within the Division, and for obtaining technical consulting on specific aspects of the review from groups outside the Division of Reactor Licensing.

In the course of the review of the material submitted, a number of meetings were held with representatives of the applicant to discuss the proposed plant. As a consequence, additional information was requested of the applicant which was provided in certain of the amendments. A chronology of the review process is attached as Appendix A to this report. Appendices C through G include reports from our consultants on meteorology, cooling water dilution studies, geology, seismology, structural design, and radiological monitoring, respectively.

We also have attached to this safety evaluation for Unit No. 2 the reports of our consultants who advised us during the safety review of the Diablo Canyon Unit No. 1 facility.

The Commission's Advisory Committee on Reactor Safeguards (ACRS), in consideration of this project, has met with both the applicant and the staff. A copy of its report to the Commission on Diablo Canyon Unit 2 is attached as Appendix B.

The evaluation of the proposed Diablo Canyon Unit 2 at the construction permit stage is only the first stage of the continuing review of the design, construction, and operation of the plant. Prior to issuance of an operating license, we will review the final design to determine if all the Commission's safety requirements have been met. The facility would then be operated in accordance with the terms of the operating license and the Commission's regulations, under continued surveillance of the Commission's regulatory staff.

The issues to be considered, and on which the findings must be made by an Atomic Safety and Licensing Board before the requested license may be issued, are set forth in the Notice of Hearing issued by the Commission in this proceeding and published in the Federal Register.

2.0 SITE

2.1 Description

The site for the proposed nuclear plant is adjacent to the Pacific Ocean in San Luis Obispo County, California. The site consists of approximately 750 acres near the mouth of Diablo Canyon Creek. The 585-acre portion south of the creek is leased to the company for a term of 99 years with an option to renew for an additional 99 years. The 165 acres north of the creek is owned by PG&E in fee. The minimum exclusion area distance from the reactor to the nearest site boundary on land will be one-half mile.

The nearest residence from the site is approximately 1-3/4 miles and the area out to a distance of about 6 miles is sparsely populated. The

six-mile distance was selected by the applicant as the low population zone radius. The nearest population center is San Luis Obispo. Its nearest boundary from the site is ten miles, which is considered to be the population center distance.

2.2 Meteorology

For evaluation of accidents the applicant has chosen diffusion parameters which correspond to a Pasquill Category F meteorlogical condition with one meter per second wind speed for short term releases, and meteorological conditions corresponding to a distribution of Pasquill Category C, D, and F for long term releases. The Environmental Science Services Administration (ESSA) has evaluated the meteorological assumptions and has concluded that the assumptions chosen by the applicant are conservative. This conclusion is based on a review and analysis of the meteorological data taken by the applicant at the site. The data, presented in Amendment 6 in response to the questions raised by ESSA in its August 11, 1969 comments, show that more than 95% of the time the actual meteorological conditions at the site are more favorable for atmospheric dispersion in the on-shore direction than those assumed for accident evaluations. The ESSA reports are attached in Appendix C to this report. The applicant's meteorological program includes meteorological measurements from a 250-foot tower near the plant location, from a 100-foot tower at the top of a 914-foot hill on the site, and at four other locations. We conclude that the meteorological program is adequate to provide a basis for the development of a gaseous radioactive release limit and to confirm the conservatism of diffusion parameters used in the analysis of potential accidental releases.

2.3 Geology and Hydrology

The geologic features of the plant site were presented in the Preliminary Safety Analysis Report.

The U. S. Geological Survey (USGS) has reviewed the application and other available literature, and has examined the exploratory trenches at the site. It has concluded that the applicant's analysis appears to be carefully derived and to present an adequate appraisal of those aspects of the geology which would be pertinent to an engineering evaluation of the site. The report of the U. S. Geological Survey (USGS) is attached as Appendix D of this evaluation. Our structural consultants, N. M. Newmark, W. J. Hall, and A. J. Hendron, Jr., have recommended that analyses be performed of the stability of slopes that might present a hazard to the Unit. Their report is attached as Appendices F-1 and F-2. We concur in this recommendation. Based upon these reports, with the understanding that stability analyses will be carried out, we have concluded that the geology of the site presents no unusual engineering problems for the construction of this nuclear facility.

The hydrological characteristics of the site were previously reviewed by the USGS for Unit 1. It was concluded that the reactor location would not be affected by floods of Diablo Canyon Creek, the only developed drainage nearby, except perhaps for part of the switchyard. It was noted that there were no reports of ground water developmentsin the vicinity of the site and concluded that it does not appear that the reactor would affect fresh water resources of the site. These conclusions remain applicable for the Unit 2 facility.

2.4 Seismology

The seismic history of the Diablo Canyon area was studied by the applicant and its consultants and by the U. S. Coast and Geodetic Survey for the Unit 1 site. The results of this study apply to the adjacent Unit 2 site. In summary, the applicant concluded that the following four possible types of earthquakes would result in maximum accelerations at the site:

Earthquake A: Magnitude 8-1/2 along the San Andreas Fault 48 miles from the site, resulting in a ground acceleration of 0.10 g at the site.

Earthquake B: Magnitude 7-1/4 along the Nacimiento Fault 20 miles from the site, resulting in a ground acceleration of 0.12 g at the site.

Earthquake C: Magnitude 7-1/2 along the off-shore extension of the Santa Ynez Fault 50 miles from the site, resulting in a ground acceleration of 0.05 g at the site.

Earthquake D: Magnitude 6-3/4 aftershock at the site associated with Earthquake A which results in a ground accleration of 0.20 g at the site.

For design pruposes, the applicant proposes to use both an envelope of the B and D response spectra and the B and D spectra separately. The operational basis earthquake (OBE) would encompass Earthquake B with a horizontal accleration of 0.15 g and Earthquake D with a horizontal accleration of 0.20 g. The use of two earthquake response spectra for the OBE was selected by the applicant because the frequencies of ground motion for the two earthquakes are different as a consequence of unequal attenuation due to earthquake location. The applicant further states the design will be reviewed using 0.30 g and 0.40 g for the design basis earthquake for this site. The applicant has recommended, and we agree, that the same acceleration

values be employed for the Unit 2 design.

2.5 Oceanography

Condenser cooling water and auxiliary salt water cooling for the plant will be taken from the Pacific Ocean and returned through an outfall via Diablo Cove. All of the Class I structures and equipment are located 60 or more feet above mean sea level with the exception of the auxiliary salt water intake structure. The peak tsunami wave height, coincident with peak storm and high tide run-up, is approximately 18 feet above the mean lower low water (MLLW). The applicant will provide protection for this equipment to an elevation of 30 feet above MLLW.

The maximum draw-down due to a tsunami coincident with low tide is nine feet below MLLW. The bottom elevation of the intake structure is 28.9 feet below MLLW and the auxiliary salt water pumps are designed to operate with the water level down to 17.4 feet below MLLW. Therefore, the ocean down to this level provides a reservoir for the auxiliary salt water system during a tsunami downsurge.

Liquid wastes containing small amounts of radioactivity will be diluted in the plant circulating water and discharged to the ocean at concentrations well within the limits of 10 CFR Part 20. Wastes will be discharged on a batch basis and will be monitored and controlled to assure compliance with 10 CFR Part 20. The applicant has also performed rhodamine dye tests to estimate the dilution capacity of the coastal ocean waters. The comments of our consultant, USGS, on the applicability of these tests are presented in Appendix D-2. As a further check, the applicant's environmental monitoring program includes sampling of ocean water, sediments, and marine life

in the plant vicinity during plant operation.

2.6 Environmental Monitoring

The applicant proposes a preoperational environmental monitoring program which will begin about two years before operation of Unit 1. Background radiation will be measured, and samples will be taken of air particulates, bovine thyroids, milk and vegetables. Sampling of marine life will include abalone, clams, rockfish, salmon, mussels, barnacles and bull kelp (seaweed). Bottom sediments and seawater samples will also be included. After completion of the two-year program, it will be reviewed, modified where experience indicates that changes should be made, and continued during plant operation.

The U. S. Fish and Wildlife Service, Department of the Interior, has reviewed the proposed environmental monitoring program and concluded that is is essentially the same as the program for Unit 1, and that comments contained in its letters of June 23, 1967, and January 3, 1968, apply to the Unit 2 program. In its letter of January 3, 1968, F&W recommended that the environmental monitoring program proposed for Unit 1 include sampling of seawater and sediments. These have been included in the proposed program for Unit 2. Comments of the Fish and Wildlife Service are attached as Appendices G-1, G-2, and G-3.

2.7 Conclusions

On the basis of our review we conclude the site is acceptable for the proposed nuclear facility.

3.0 NUCLEAR STEAM SYSTEM DESIGN

The nuclear steam supply system consists of a light-water-moderated pressurized water reactor (PWR) which transfers reactor heat to four steam generators. It is similar in design to other PWR systems that have been granted construction permits and operating licenses. There have been evolutionary changes in the design of Diablo Canyon Unit 2 since issuance of the construction permit for Unit 1 and the applicant states that these changes will also be made in the design of the previous unit. The major design modifications in the nuclear steam supply system involve the emergency core cooling systems which are discussed in Section 5.1 of this report. One other major design change is the change of control rod absorber material.

The control rods for Unit 2 will contain boron carbide neutron absorbing material rather than silver-indium-cadmium as originally proposed for Unit 1. The nuclear control characteristics of the reactor are not affected since the material substitution does not significantly change the rod worths. We presented questions to the applicant concerning the potential for control rod swelling and the applicant's response is contained in Amendment No. 5. We have concluded that the change in absorber material is acceptable.

For Diablo Canyon Unit 2, we have further reviewed the applicant's proposed instrumentation to assure that the power distribution is adequately controlled. The applicant has proposed that the four external flux monitors will detect abnormal power patterns. The in-core instrumentation for determining the power distribution in the Diablo Canyon core, as presently

proposed, consists of six traveling flux probes which may traverse any of 58 thimble locations, and 65 thermocouples located in guide tubes at the exit flow ends of the fuel assemblies. The in-core flux probes are not designed to operate in the core at full power for more than a few months. The applicant believes that the planned test programs (primarily at the Ginna and Indian Point 2 plants) will adequately demonstrate prior to operation of Diablo Unit 2 the capability of the external long ion chambers to measure core power distributions at any time during core life. It remains to be demonstrated that the proposed external monitors can detect incore power distributions and anomalies with adequate sensitivity to assure that no loss of safety margins can occur. In the event this cannot be shown, fixed in-core monitors will be required to assist the operator in positioning the part length rods for proper power distribution control.

In this regard, the applicant has agreed that the in-core monitors will be used with frequent flux scans, or other monitors could be semi-permanently installed in the existing in-core thimbles for this purpose. We will review this aspect again at the operating license stage of our review.

On the basis of our review of the information provided by the applicant and on our previous reviews of similar PWR systems, we have concluded that the proposed nuclear steam supply system, including the nuclear design, the thermal and hydraulic design, the mechanical design, the plans for the instrumentation and control systems design and the secondary system design is acceptable.

4.0 CONTAINMENT

4.1 Description

The proposed containment vessel is a reinforced concrete structure with a steel liner which encloses the reactor and reactor coolant system. The internal concrete structures will be designed to provide missile protection for the primary coolant systems in the event of failures of valves, including valve stems and bonnets, instrument thimbles, closure bolts, and complete control rod drive mechanisms. The design will also include provision to withstand forces associated with a double-ended rupture of a main coolant pipe.

Further detail on the containment design is presented in following sections.

4.2 Containment Loading

Factored loads for the design of the containment structure have been proposed which combine dead loads, live loads, pressure loads, temperature loads and earthquake loads (or wind loads if greater than the earthquake loads). The containment will be designed such that the most restrictive load combination for each particular region of the containment results in average stresses not greater than the yield point.

The factored loads indicated that the containment will have the capacity to withstand loadings, as follows:

(a) at least 50 percent greater than those calculated for the loss-of-coolant accident,

(b) at least 25 percent greater than those calculated for the loss-of-coolant accident coincident with the operational basis earthquake,

(c) at least as great as those calculated for the loss-of-coolant accident coincident with an earthquake twice the magnitude of the operational basis earthquake with no loss of function.

The design pressure for the containment is 47 psig. The applicant has performed a parametric study of the containment pressure as a function of time using various assumptions for the course of the accident. The engineered safety features assumed to be operable in these studies were one of two containment spray systems and three of five containment fan-cooler units. The design basis case assumes a primary system break size of about three square feet which is calculated to result in transfer of the maximum available net energy from the core internals and vessel to the coolant and in the highest peak containment pressure. This design basis accident results in a peak containment pressure of 39.6 psig.

Analyses were also performed by the applicant to demonstrate the adequacy of the containment design pressure by including additional energy transferred from stored energy in the components and from reactor decay heat, and by including metal-water reactions. Inclusion of a metal-water reaction which involves about 32 percent of the Zircaloy within 1000 seconds causes a peak containment pressure of 45.7 psig, which is less than the design pressure.

Earthquake loadings will be computed on the basis that the vertical acceleration values are two-thirds of the horizontal ground acceleration values with the effects of the horizontal and vertical loadings combined on the assumption that they act simultaneously. The operational basis earthquake and the design basis earthquake are defined in terms of acceleration in Section 2 of this report. Our consultants have reviewed the seismic aspects of the containment design and their conclusions, with which we concur, are included in Appendix F to this report.

4.3 Structural Design Details

The containment reinforcing consists of hoop bars and helical bars extending the full height of the wall at the exterior face, and continuing over the dome to meet the reinforcing on the other side. The diagonal reinforcing at the inside face of the wall reaches an elevation of approximately 83'-0". This arrangement permits the main reinforcing at the exterior face to be placed with no need for end anchorage points in zones of biaxial tension in the structure. The applicant's criteria for splice stagger and user testing of rebar are adequate. The ASTM A615 GR60 reinforcing bars will be limited in carbon and manganese content to assure better fusion welding and bending properties, although cadweld splices will be used where possible. The applicant has agreed to utilize cadweld splice testing criteria which conform to our requirements.

A hinge design at the base of the containment wall structure permits the wall to rotate outward at the base while still preventing radial expansion of the wall at the base. We and our dynamic design consultants have reviewed the hinge concept and design criteria and find them to be acceptable.

The 18.5 foot diameter equipment hatch and the 9.0 foot diameter personnel hatch are framed by 2-1/2 inch thick A-36 structural steel rings which transfer the loads around the openings. The ASME Boiler and Pressure Vessel Code Section III will be followed in design, fabrication, and heat treating of the rings. The applicant has stated that proper consideration will be given during design to the fact that the ring will not be planar. Connection of reinforcing to the ring will be through a rod with a Cadweld sleeve on the rebar end and a threaded connection at the steel ring. We find the design criteria for these rings acceptable.

The steel liner for the containment will conform to ASTM A516, Grade 70, and will be fabricated in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII and IX. Since the design criteria for the liner result in a factor of safety against buckling of 1.17, we find the design approach acceptable.

The dynamic analysis methods and assumptions used in evaluating the design of the containment have been reviewed by us and our dynamic design consultants and found to be acceptable.

.4 Testing and Surveillance

All liner plate welds will be covered by channels, strength tested at 54 psig for fifteen minutes, then leak tested at 47 psig. The acceptable leak rate will be no greater than 0.1% of the containment free volume per day. The wall and dome liner plate welds will be radiographed for the first ten

feet made by each welder for each position, and then at least ten percent of the weld thereafter will be radiographed.

The containment structural proof tests will be conducted at 35, 40, 47, 50, and 54 psig pressure. The liner strains will be measured by means of electrical strain gauges. Radial and longitudinal growth of the cylinder will be measured through visual targets on the structure. Radial growth at the base will be monitored by linear motion transducers. Visual checks on specified areas will be carried out to verify crack patterns and sizes.

If major maintenance or modifications of the containment building are made, it will be possible to pressure test it to 54 psig. The applicant has stated that to perform this on Unit 2, an evaluation would have to be made of the necessity to close down Unit 1 and evacuate personnel during the test, but if a 54 psig test should be required in the future, it would be possible to conduct it.

Periodic testing of the containment as proposed by the applicant will consist of pressurizing the double-walled penetrations and weld seam channels without an integrated leakage test. We will require the periodic performance of an integrated leak test at a pressure and frequency to be developed at the operating license review stage.

Based on our review we conclude that the applicant's proposed testing and surveillance programs are acceptable for the construction permit stage of review. Requirements for periodic integrated leakage testing of the containment will be considered further during the operating license stage of our review.

4.5 Conclusion

We conclude that the Diablo Canyon Nuclear Unit 2 containment and Class I structures can be adequately designed, constructed, tested and utilized under the criteria presented by the applicant, and are acceptable.

5.0 ENGINEERED SAFETY FEATURES

5.1 Emergency Core Cooling Systems

The emergency core cooling system (ECCS) for this plant is an improved version of those previously reviewed for other 4-loop Westinghouse PWR's. It consists of (a) part of the reactor charging system, (b) a high pressure coolant injection and recirculation subsystem (HPS), (c) two low pressure coolant injection and recirculation subsystems (LPS), and (d) an accumulator subsystem.

The minimum ECCS capability (assumed in design basis accident evaluations) is 3 accumulators, one low pressure pump, one high pressure pump, and one centrifugal charging pump, all delivering at rated capacity and powered by the diesel generators. For long term coolant recirculation, the minimum ECCS capability can be provided even in the event of a failure of any single active or passive component.

All of the ECC subsystems can accomplish their functions when operating on normal or emergency (onsite) power. If one of three diesels fails to start following a LOCA and loss-of-offsite power, the minimum ECCS will be providing coolant (assuming no further component failures) within 30 seconds following a safety injection signal.

The applicant presented ECCS performance analyses based on computer codes developed by Westinghouse. For large and intermediate sized breaks the performance of the minimum ECCS is identical to that of other four loop, 1000 MWe class PWR's. The calculated peak temperatures are below the range for accelerated Zircaloy-water reactions (i.e., less than 2200° F) and the total clad-water reaction for any break size is much less than one percent of the total fuel clad mass. Furthermore, only a samll percent ($<$ 1.5%) of the

total clad mass ever exceeds 1800° F and then only for a brief time (< 30 sec.). We have reviewed Argonne National Laboratory as well as Westinghouse regarding potential thermal shock shattering of oxidized Zircaloy cladding, and conclude that the core heat transfer geometry should not be significantly altered by thermal shock upon quenching because the time at elevated temperatures would be so short.

On the basis of our review, we conclude that the Diablo Canyon Unit 2 ECCS will (a) limit the peak clad temperature to less than the melting temperature, (b) limit the fuel clad-water reaction to less than one percent of the toal clad mass, (c) terminate the clad temperature transient before the geometry necessary for core cooling is lost and before the clad is so embrittled as to fail upon quenching, (d) reduce the core temperature and then maintain core and coolant temperature levels in the sub-cooled condition until accident recovery operations can be accomplished, and (e) provide rapid boron injection to minimize the power transient for a steam line break and to preclude a return to power from zero power for the steam line break equivalent to opening of a safety valve. The ECCS will provide this protection for all pipe breaks up to and including the double-ended rupture of the largest reactor coolant pipe.

5.2 Containment Spray System

The applicant has provided a containment spray system containing redundant active components whose function is to spray cool water into the containment atmosphere in the event of a loss-of-coolant accident in order to remove heat. The system is designed to Class I seismic requirements with the vessel meeting ASME Section III requirements and the piping meeting USAS B31.1

requirements. All of the active system components are located outside of containment; the spray headers inside containment are protected from missiles originating within the shield.

The system includes the capability for adding NaOH solution to the spray in order to effect iodine removal from the containment atmosphere. The elemental iodine removal characteristics of NaOH are well established and its compatibility characteristics with various materials have been extensively investigated. The applicant is presently modifying the NaOH addition system in order to provide the operator with additional information concerning its status during operation. This modification will be reviewed at the operating license stage.

Based on our review of the containment spray system, we conclude that the design is acceptable.

5.3 Containment Isolation System

The containment isolation system is intended to close the various piping systems which penetrate the containment to preclude the escape of radioactivity to the outside environment in the event of a loss-of-coolant accident. The isolation system provides a minimum of two barriers between the outside environment and the containment atmosphere, the reactor coolant system, or any closed systems inside the containment which may be vulnerable to the accident forces. All valves and equipment which are intended to be isolation barriers are protected against potential missiles and water jets and are designed to Class I seismic requirements.

For those lines which must be isolated immediately following an accident, automatic trip signals to the valves are provided; no reliance is placed on manual operation. Other lines which must remain in service subsequent to an accident for safety reasons are provided with at least one remote-manual valve.

Isolation valves are automatically tripped to their closed position by one of two separate containment isolation signals. The majority of the valves which are in the "non-essential" process lines are tripped in conjunction with automatic actuation of the safety injection system. These are pipelines whose isolation will not increase the potential for damage of containment equipment. The remaining valves, located in "essential" process lines, are tripped upon actuation of the containment spray system. The essential lines (i.e., those providing cooling water and seal water flow to the reactor coolant pumps) are not interrupted unless absolutely necessary. Both the automatic isolation valves and remote-manual valves are operable from the control room area. Position indication for each valve is also provided on the control room panels.

The isolation valves are designed to close upon loss of control power or air. Also, the instrumentation and control circuits are redundant in that a single failure will not prevent containment isolation. In addition, provisions are included to permit periodic testing of the leak tightness and functioning of the isolation valves.

We have reviewed the design of the containment isolation system and the applicable design criteria and have found them to be acceptable.

5.4 Hydrogen Control

In Amendment 3 to the PSAR, the applicant discussed the potential for accumulation of hydrogen in the containment following a loss-of-coolant accident and a proposed method for limiting the concentration to preclude an explosion hazard. The various radiolytic and chemical mechanisms for hydrogen generation were considered and the applicant concluded that

the concentration of hydrogen would reach the lower flammability limit in the containment in approximately 130 days following the accident. At this time, the applicant would propose to vent the containment gases to the atmosphere through filters and a charcoal bed. The applicant states that by controlling the venting operation over a 30-day period, offsite doses from I-131 and Kr-85 could be maintained below the limits of 10 CFR 20, if averaged over a one-year period.

Using parameters for hydrogen production which are somewhat more conservative than those assumed by the applicant, we calculated that the lower flammability limit would be attained in less than 40 days. For this case, the purging operation could result in activity concentration levels offsite which exceed 10 CFR 20 limits; however, additional capability for filtering of containment effluent, including charcoal beds, would reduce the I-131 concentration levels. We have calculated that, assuming 90% filter efficiency for iodine removal, doses at the site boundary would be about 0.8 rem whole body and about 8.5 rem to the thyroid if the entire contents of the containment were vented over a 30-day period.

We are continuing our evaluation of the potential for hydrogen accumulation in large water-cooled power reactors and of systems for post-accident hydrogen control. On the basis of our review to date, we have concluded that the venting approach is technically feasible and will limit offsite doses. However, continued effort should be pursued in developing means in addition to containment venting, such as flame and catalytic recombiners, hot surface combustion or the use of chemical additives in the spray solution for scavenging hydrogen. We will continue to work with the applicant on this problem area during construction.

6.0 DESIGN OF CLASS I STRUCTURES FOR SEISMIC AND ACCIDENT LOADINGS

6.1 General

Those structures, systems and components of the nuclear plant which are important to nuclear safety, i.e., failure of which might cause or increase the severity of a loss-of-coolant accident or result in the release of excessive amounts of radioactivity, are termed Class I. Those components which are not essential to the safe shutdown of the reactor and failure of which would not result in the release of substantial amounts of radioactivity are considered Class II. Those structures and components not related to reactor operation are Class III. The subsequent discussion on design criteria contained in this section relates to the Class I structures and components.

6.2 Earthquake Response Spectra and Damping

The magnitude of ground acceleration for earthquakes at this site, including the review of this aspect by the U. S. Coast and Geodetic Survey for Unit 1, was discussed previously in Section 2. The response spectra for the 0.15 g ground acceleration earthquake ("far away") and that for the 0.20 g ground acceleration earthquake ("close-by") were presented in the application. Together these response spectra represent the operational basis earthquake. A larger 0.40 g ground acceleration earthquake response spectrum is also contained in the application, which represents the design basis earthquake.

The damping values to be used in the design are presented in the PSAR. These values compare favorably with values used in the design of other facilities.

During detailed design, vibrational response characteristics will be calculated for the assumed earthquakes. We agree with the ACRS that consideration should be given to obtaining experimental verification of the anticipated behavior of components and instrumentation to the extent practicable.

6.3 Design Loading and Stress-Strain Limits

This section pertains to the design loading and limits for Class I structures, systems and components including the pressure vessel and piping, but excluding the containment structure. (The design loading and stress limitations for the containment structure were described in Section 4.2). The requirement for these items is that they should be designed to withstand:

(a) Load combinations including normal design loads and operational basis earthquake loads within normal working stress or deflection limits.

(b) Load combinations including design basis earthquake loads and applicable design basis accident loads, without loss of function of the specific structure, system, or component.

We have reviewed these requirements and we consider these loading combinations and limits to be both realistic and satisfactory. The proposed stress or deformation limits for specific components are discussed in more detail below.

Reactor Vessel Internals

For normal design loads of mechanical, hydraulic, and thermal origin, including anticipated plant transients and the operational basis earthquake, the reactor internals will be designed to the stress limit criteria of the ASME Boiler and Pressure Vessel Code Section III.

The reactor internals will also be designed to withstand the concurrent loads resulting from the loss-of-coolant accident and the design basis earthquake. The direct primary stresses under this combined loading will not exceed stresses corresponding to 20 percent of uniform strain at temperature. Allowable deflections will be limited to about 50 percent of the design loss-of-function deflections for the specific components. Preliminary calculated deflections are smaller than even these limits.

We find that the stress and deflection limits discussed above provide an adequate margin of safety.

Vessels, Piping and Supports

The ASME Section III and B31.1 Code stress limits for vessels, piping and supports as proposed by the applicant are considered satisfactory for type (a) load combinations.

For the type (b) load combination, the allowable extent of plastic deformation can be larger than that associated with the ASME Section III stress limits. Further, the limits on strain defined in Appendix F of the PSAR will assure no loss of function. We conclude that the design provides an adequate margin of safety.

6.4 Conclusion

On the basis of the information presented in the application and the analyses of the design by our consultants Newmark and Hall as attached as Appendix F, we have concluded that the proposed design criteria for Class I structures systems and components provide an adequate margin of safety to withstand the selected combined loading conditions.

7.0 INSERVICE INSPECTABILITY AND SURVEILLANCE

In Amendment 1 to the PSAR, the applicant has indicated that the design of the reactor coolant system and its components makes provision for virtually complete compliance with the draft ASME Code for Inservice Inspection of Nuclear Reactor Coolant Systems dated October 1968. The applicant is currently formulating plans for inservice inspection and will submit them for our review prior to the application for an operating license. The plans are intended to reflect the requirements of the latest accepted revision of the inservice inspection code to the extent practical.

Visual and/or non-destructive inspection techniques can be applied on a conventional basis to all primary system components (including the reactor coolant pump flywheel) with the exception of the reactor vessel because of high radiation levels and remote underwater inaccessibility. In some areas, the inspection must be limited to external or internal surfaces or cannot be performed in shield penetration areas because of accessibility difficulties. For the reactor vessel, however, volumetric inspections must await the availability of non-destructive testing techniques suitable for areas of high radiation. In anticipation of the development of these techniques, the applicant will perform ultrasonic test mapping of selected areas of the vessel during fabrication.

On the basis of our review of the applicant's approach to inservice inspection, we conclude that an acceptable program will be developed. We will review the applicant's program as it is developed during construction.

The applicant has included provisions for a surveillance program to monitor the extent of radiation damage to which the reactor vessel is exposed. The program provides for eight capsules containing a minimum of 351 specimens for tensile, Charpy V-notch, and wedge opening loading testing. A tentative schedule for capsule removal during the operational life of the plant is provided. The specimens are located so that they experience higher irradiation than the vessel wall thereby producing data representative of the vessel at a later time in life. Based upon our review, we conclude that the applicant's proposed program is acceptable.

8.0 ELECTRIC POWER SYSTEMS

Offsite Power

Diablo Canyon Units 1 and 2 will be interconnected to the PG&E systems through 500 kV and 230 kV circuits. Power from both Units 1 and 2 feed the 500 kV switchyard. Three transmission lines emanate from the 500 kV switchyard on two separate rights-of-way. The 230 kV switchyard interconnects the plant to the PG&E system by means of two circuits, each on a separate right-of-way.

Redundant independent sources of offsite power provide power to the 4160 volt buses which feed the engineered safety features upon loss of the normal unit supply. One source is derived from the standby-startup transformer connected to the 230 kV switchyard. The second source is made available by backfeeding through the main transformer from the 500 kV switchyard. This is made possible by a manually initiated motor-operated disconnecting link in the generator's main leads. This disconnect link is operated from the control room and is interlocked to prevent opening under load. The applicant states that this source of power could be made available in approximately 30 seconds.

We conclude that sufficient redundant and independent sources of offsite power are provided to reasonably ensure that no single failure will cause the loss of offsite power to the engineered safety features.

Onsite Power

The engineered safety feature loads are divided among three 4160 volt buses such that operation of any two will supply the minimum safety requirements. Three 2500 kW diesel generators are provided. Two are

exclusively assigned to two of the above mentioned buses, and the third diesel generator is capable of being connected to either the third bus of Unit 2 or to a similar bus in Unit 1. The fuel supply is sufficient to operate the diesels required for accident loads in one unit and safety shutdown loads in the second unit for a period of seven days. The applicant has made provision for quick resupply from nearby sources.

Two d-c systems are provided for control and protection instrumentation, annunciation, and emergency lighting and lubricating systems. One is a 250/125 volt three-wire ungrounded system with two 125 volt buses, and one 250 volt bus for turbine generator emergency motors. The other system is a separate 125 volt two-wire ungrounded system supplying a 125 volt bus. Each of these buses has its own battery and its own redundant battery chargers. Sufficient battery charger capacity is provided to carry the normal continuous load and to recharge the batteries with any one charger out of service. The batteries are sized for two-hour operation of all vital loads. Each battery is located in a separate room in a Seismic Class I area. The losss of any one battery will not preclude the operation of the minimum required engineered safety features.

The 120 volt a-c vital instrument bus system supplies power for control and protection system instrumentation and alarms. These loads are divided into four groups. Each group is served by a separate inverter supplied from the station batteries. A normally open standby connection between the inverter fed buses is available for use when an inverter is out of service.

Based on our review, we conclude that all relevant aspects of the onsite power system comply with our criteria, and that the system design is acceptable.

9.0 RADIOACTIVE WASTE CONTROL

Liquid wastes are collected during plant operation and pumped to the waste holdup tanks for batch processing by the Waste Disposal System. Liquid processing will include filtration and evaporation as necessary. Liquid releases to the condenser discharge tunnel will be in conformance with 10 CFR Part 20 limits and will be continuously monitored by a radiation monitor. High radiation from the monitors will automatically close a discharge valve in the liquid waste disposal system. Prior to release of any batch of liquid, samples will be analyzed to determine the type and amount of activity in the batch to ensure conformance with release limits which will be established at the operating license review stage.

Solid radioactive wastes from plant operation will be temporarily stored on site and shipped from the site in containers approved for that purpose.

Gaseous radioactive wastes from the chemical and volume control system, various cover gases and vents will be collected and compressed into Gas Decay Tanks. After a suitable decay period, the contents of a tank will be sampled to determine its activity and released to the vent pipe. A radiation monitor is also provided in this discharge line and, should the radiation level become high during the release, will automatically close a valve in the discharge line from these tanks. The limits for gaseous release in accordance with 10 CFR Part 20 will be established at the operating license review stage.

We conclude that the waste disposal system proposed by the applicant will effectively control the discharge of radioactive waste generated on the site in accordance with AEC regulations and that it is acceptable.

10.0 ACCIDENT ANALYSIS

10.1 Operating Transients

A number of plant operating transients were considered by the applicant including rod withdrawal during startup and from power, boron dilution, loss-of-coolant flow, loss of electrical load, and loss of offsite AC power, in order to assess the safety margins of the plant design. The criterion for detailed design of the reactor control and protection system is to be able to automatically take corrective action to copy with any of these transients. Preliminary analyses as presented in the application will be recalculated during detailed plant design to verify that transients are within the capabilities of the reactor control and protection systems. Based on our evaluation of the information submitted and on the results of evaluations of other PWR designs at the operating license stage, we conclude that anticipated transients can be terminated with adequate margin.

10.2 Accident Evaluation

Potential accidents which could result in radioactive releases to the environment have been analyzed by the applicant. We have evaluated these accidents and the engineered safety features provided to limit the potential exposures. During the course of the detailed design the applicant will continue to reassess the potential consequences of the various accidents,

and we will evaluate the final design at the operating review stage. We have evaluated the potential consequences of these accidents and all of the resulting doses, with the exception of the postulated fuel handling accident, are well below the 10 CFR 100 guideline dose levels at the available exclusion zone radius (0.5 mile) and the low population zone radius (6 miles). The more significant of these accidents are discussed in greater detail in the following sections.

10.2.1 Loss-of-Coolant Accident

The design basis loss-of-coolant accident for Diablo Canyon Unit 2 is similar to that used for other PWR's in that a double-ended break in the largest pipe (9.2 ft^2) in the reactor coolant system was assumed. As discussed in Section 5.1, the emergency core cooling systems are designed to limit fuel cladding temperatures to well below melting temperature. Although the basis for sizing the emergency core cooling system is to limit fission product release from the fuel, we nevertheless require that the containment and its associated engineered safety features shall be capable of limiting potential doses in conformance to 10 CFR Part 100 guidelines assuming significant releases of fission products from the fuel. We analyzed the consequences of this accident assuming that 100% of the core noble gases, 25% of the core iodines and 1% of the core solids are released to the primary containment atmosphere. Based on our evaluation of the containment spray system, we assumed an iodine removal time constant of 3.7 per hour for the sodium hydroxide sprays. We also assumed that 10% of the iodine would be in a chemical form not removable by the spray system and that the containment leaked at the design basis rate

of 0.1% per day for the first day and one half that value for the remaining 29 days.

We assumed a ground release using Pasquill Type F meteorology and 1 m/sec wind speed for the first 8-hour period after an accident. The same meteorological conditions with a uniform dispersion into a 22-1/2° sector were used for the 8 to 24 hour period and the stability, wind speed, and wind direction were varied for the 1 day to 30 day period. We calculated potential doses for the loss-of-coolant accident at the exclusion distance for a two-hour period of 4 rem and 190 rem for the whole body and thyroid, respectively. The corresponding 30-day potential doses at the low population zone boundary were less than 1 rem whole body and 26 rem to the thyroid.

10.2.2 Fuel Handling Accident

In our analysis of this accident we assumed that all of the fuel rods in a dropped fuel assembly (204) were damaged, that 10% of the iodines and 20% of the noble gases contained in the rods were released to the refueling water, that 10% of the iodines and all of the noble gases that were released to the water were released to the building atmosphere, that the accident occurred 100 hours after shutdown of the reactor, and that the dropped fuel assembly had previously operated at a power density 80% higher than the average core fuel assembly. Using meteorological assumptions of Pasquill Type F with a wind speed of 1 m/sec, we calculated potential doses at the exclusion radius of 4 rem whole body and 1575 rem to the thyroid.

The applicant has calculated a thyroid dose of 0.3 rem as a result of this accident using less conservative assumptions than those presented above. The differences between the applicant's calculated value and ours results from the different values assumed for iodine inventory in the rods which are released to the water; the fraction of iodines retained by the pool water; and the number of rods damaged as a result of the dropped fuel assembly.

We have discussed the above differences with the applicant and in Amendment 5 he indicated that he plans to conduct further experimental and analytical studies of the problem in order to justify his less conservative assumptions for the accident analysis. The applicant has further stated that, in the event his study results are not acceptable, he will provide additional iodine fixing equipment in the Fuel Handling Building ventilation system. We have concluded that the inclusion of such equipment could provide the dose reduction factor necessary to make the consequences of the fuel handling accident well within 10 CFR 100 guidelines. The applicant's results and the possible need for additional iodine fixing equipment will be evaluated during the operating license review.

10.2.3 Steam Line Break Accident

In our evaluation of this accident we assume that a break occurs in the steam piping between the steam generator and the turbine, resulting in the loss of all secondary water in one steam generator. The break is assumed to occur at zero power when the steam pressure is the maximum.

The resultant contraction of reactor coolant system water is characteristic of the beginning of a loss-of-coolant accident and, for large steam line breaks, causes actuation of the ECCS to limit the transient by boron injection.

For the calculation of potential doses, we assume that the break occurs while a primary to secondary leak of 10 gpm exists, the reactor coolant contains equilibrium activity corresponding to 1% fuel failures, and a continuous steam generator blowdown of 10 gpm is occurring. It is further assumed that the entire iodine inventory in the secondary side of the steam generators is released. The resultant calculated 2-hour thyroid dose at the exclusion radius is 39 rem, well within the guidelines of 10 CFR 100.

10.2.4 Steam Generator Tube Rupture

We have evaluated the potential consequences of a postulated double-ended rupture of a steam generator tube. We assumed that reactor coolant, amounting to 80,000 lbs. and containing equilibrium activity due to 1% failed fuel, would be released directly to the environment through the secondary safety valve over a five hour period. This release would result from a simultaneous loss-of-offsite power permitting no bypass to flow to the main condenser. The leakage of reactor coolant is assumed to persist until the system pressure is reduced to below 1100 psia, permitting isolation of the faulty steam generator; in practice this should occur in less than one-half hour . No core damage is anticipated for this accident which

results in actuation of the safety injection system should no operator action occur. On the basis of the above assumptions, we calculated a thyroid dose of 35 rem at the exclusion radius - well within the 10 CFR 100 guidelines.

11.0 QUALITY ASSURANCE

In Amendments 4 and 5 to the PSAR, the applicant has described the Quality Assurance Program (QAP) which it proposes to utilize for the design and construction of the Diablo Canyon Unit 2 reactor plant. The applicant states that its QAP is intended to be responsive to the recently published AEC quality assurance criteria.

PG&E performs its own architect-engineering. The only major contractor is Westinghouse, the supplier of the nuclear steam supply system. Within the applicant's engineering department a new group has been established separate from the design group with responsibility for a continuing review of the QAP and for reporting on its adequacy to PG&E management. This new group is headed by the Director, Quality Engineering, who is directly responsible, as is the Project Engineer for Unit 2, to the Vice-President of Engineering. This arrangement is intended to provide a quality assurance activity which is independent of the design and construction activities. The Director, Quality Engineering will have a staff of engineers and specialists who will audit quality related activities during design and construction. This organizational arrangement satisfies our requirements.

In his description of the QAP, the applicant has presented its approach to each of the QA criteria specified in 10 CFR 50 Appendix B (Proposed). The program requires that planned and documented actions be applied to all quality related activities which involve the structures, systems and components important to safety. All contractors, including Westinghouse, will be required to provide a QAP to the extent necessary as determined by their scope of work. Surveillance of contractor's efforts will be performed by PG&E.

Although many aspects of the applicant's QAP will require further definition, the applicant's commitments and its planned approach in each of the critical areas satisfy the criteria for the construction permit stage. During construction, we will follow the development of the details of the applicant's QAP to assure adequate implementation of current plans.

12.0 RESEARCH AND DEVELOPMENT PROGRAMS

The applicant has identified several areas in the design of the plant requiring further research and development effort. These areas include those in which additional information is required for plant operation and those which will provide added confirmation that the contemplated designs are sufficiently conservative. None of the research and development areas can be characterized as unique to Diablo Canyon Unit 2; rather, they are common to all Westinghouse PWR's and the results of these efforts as they are performed for preceding PWR's will be applicable to Unit 2. Each of the research and development areas is briefly summarized below:

12.1 Power Distribution Control

It is necessary to demonstrate the capability of out-of-core detectors to indicate axial and diametral core power distribution to permit control of xenon oscillation and to warn of misplaced control rods, to develop a control system for axial power shaping with part-length rods and for diametral power shaping if necessary with full length rods, and to verify during start-up testing that the control system satisfactorily provides power distribution control and adequate safety margins. We do not agree with the applicant that the initial two objectives have been adequately accomplished. We are not convinced that an adequate capability for determining the nuclear status within large cores such as Unit 2 has yet been demonstrated by out-of-core detectors. The applicant has made provision for the use of incore detectors should they become necessary.

Westinghouse expects to fulfill these objectives during startup testing and subsequent operation of the Ginna and Indian Point Unit 2 plants scheduled for 1969 and 1970 respectively. Final verification of the adequacy of these control systems in later units will be obtained during startup testing prior to full power operation.

12.2 Fuel Rod Burst

Efforts are underway to determine the extent of core geometry distortion which may result from a loss-of-coolant accident. This work is being performed by various industry and government groups with overall coordination by the Oak Ridge National Laboratory. Any major uncertainties regarding the extent of core geometry distortion and the capability for adequate ECCS performance should be resolved prior to operation of Diablo Unit 2.

.3 Failed Fuel Monitor

The experimental evaluation of several detector types by Westinghouse is underway at the Saxton reactor with the objective of developing a unit which will provide prompt detection of failed fuel pins. Detailed evaluation of these detector types will be completed in 1970. At the present time, a monitor in the letdown stream is included in the Unit 2 design. Determination of the adequacy of the various detector types for use in the Diablo Canyon Units will be made during the operating license review.

12.4 Burnable Poison

Experimental and analytical efforts to develop burnable poison rods have been satisfactorily completed. Verification of the design adequacy will be available from commercial operation of the Ginna plant prior to operation of Diablo Unit 2.

12.5 Fuel Development

Fuel development efforts are underway in both the Saxton and Zorita irradiation programs to determine performance margins available relative to peak power and the accommodation of overpower transients. Results from these efforts will become available beginning in late 1971 and extending through the end of 1973, prior to operation of Diablo Canyon Unit 2.

12.6 In-Core Detector

Evaluation of the lifetime of in-core detectors suitable for continuous monitoring of power distributions is proceeding in several reactor facilities such as San Onofre, Brookhaven and Western New York Nuclear Research Center. Evaluations will be completed in 1971.

12.7 Rod Bundle DNB

Experimental rod bundle departure from nucleate boiling data with non-uniform rod axial flux distributions including the effects of mixing vanes will be obtained at Columbia University under the direction of Westinghouse. Preliminary results indicate that the mixing vane provides a greater DNB heat flux than predicted.

12.8 Full Length Emergency Cooling Heat Transfer Tests

Experimental investigations are being conducted to determine more precisely the thermal behavior of a PWR core following a loss-of-coolant accident. Of particular concern is the experimental verification of the performance capabilities of proposed emergency core cooling techniques. This work, initiated in 1968, is being done by Westinghouse under contract to the AEC.

12.9 Flashing Heat Transfer

Experimental investigations of the heat transfer behavior in the core during various phases of the loss-of-coolant accident are underway. The

objective is to demonstrate that the heat transfer model assumed for the core during blowdown, uncovery, and reflooding is conservative. This work is being done by Westinghouse and the University of Michigan. The current major effort is reduction of test data.

12.10 Loss-of-Coolant Analysis

An analytical study is underway to integrate the results of the experimental programs in order to obtain a more realistic core thermal design code. Preliminary results of this effort, performed by Westinghouse, show that for the Ginna reactor peak clad temperatures of 300° to 400° F less than prior predictions are obtained for large and intermediate size breaks. A similar analysis will be performed for Diablo Canyon Unit 2 and presented in the FSAR for our review.

?.11 Blowdown Forces

A computer program has been developed to determine pressure, velocity, and force transients in order to calculate blowdown forces on reactor vessel internals. The program, BLODWN-2, has been applied to the Indian Point Unit 2 reactor with the conclusion that the calculated performance meets the design criteria. Specific analyses for Diablo Canyon Unit 2 remain to be performed and will be presented for our review in the FSAR.

12.12 Reactor Vessel Thermal Shock Analysis

Preliminary analyses have been conducted which indicate that thermal shock effects resulting from ECCS operation in the event of a Loss-of coolant accident would not be expected to cause reactor vessel failure, at least during the early years of vessel life. Further evaluations will be performed

based on analytical results and material property data being obtained from the Heavy Section Steel Technology Program at Oak Ridge National Laboratory.

Based on our review of the above research and development efforts, we have concluded that these programs should provide the required design information and assurance of design adequacy for the various areas of concern in the Diablo Canyon Unit 2 reactor plant prior to operation. The results of these programs as applied to Unit 2 will be evaluated during the operating license review; and as further information is accumulated.

13.0 TECHNICAL QUALIFICATIONS

The applicant has been active in the design, construction and operation of nuclear power plants for approximately eleven years. This experience includes the operation of the Humboldt Bay reactor and the design and construction of Diablo Canyon Unit 1. On several occasions during this time, we have reviewed the structure, responsibility and operating philosophy of the applicant's engineering organization. The Nuclear Steam System Supplier, Westinghouse Electric Corporation, has designed and constructed a number of pressurized water reactors which have been approved by the Commission. We conclude, based on these reviews, that PG&E and its contractor are technically competent to design and construct Diablo Canyon Unit 2.

14.0 CONFORMANCE TO THE GENERAL DESIGN CRITERIA

In November 1965, the Commission published its General Design Criteria for Nuclear Power Plant Construction Permits, and on July 11, 1967, published in the Federal Register its revised General Design Criteria taking into account comments received on the initial criteria and further development of the criteria by the regulatory staff. The applicant has cross-referenced the information as presented in the application with the criteria. We have evaluated the applicant and have concluded that the proposed unit conforms to the revised criteria. We will review the proposed unit at the operating license stage in light of the criteria as formulated at that time.

15.0 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Advisory Committee on Reactor Safeguards, by letter dated October 16, 1969, reported on the Diablo Canyon Unit 2 Nuclear Facility. A copy of this letter is attached as Appendix B. The letter contains a number of comments and recommendations which are listed below and noted in appropriate sections of this evaluation. These items will be resolved to our satisfaction and will be reviewed by the ACRS prior to the issuance of an operating license.

15.1 Boron Carbide Control Rods (See Section 3.0)

The Committee noted that careful attention should be paid to the possibility of control rod swelling and related consequences. Based on our review of the applicant's response to our concern, we have concluded that the use of boron carbide for the control rods is acceptable. We will examine this design aspect further during the operating license review and, if necessary, impose appropriate requirements in the Technical Specifications to provide assurance that incipient rod swelling is detected early.

15.2 Fuel Handling Accident (See Section 10.2.2)

It was observed that the applicant intends to perform experimental and analytical studies relative to the assumptions made in assuring the consequences of a fuel handling accident. Additional iodine removal equipment may be required. We will evaluate the study results during the operating license review.

15.3 In-Service Monitoring for Vibration

The Committee recommended development and implementation of means for in-service monitoring for vibration or for the presence of loose parts in the reactor coolant system. The applicant in Amendment 5 has indicated that he is evaluating the need for in-service monitoring. We will consider this aspect during the operating license review.

15.4 Common Failure Modes

The Committee expressed concern about the possibility of common mode failures which could negate the action of a protection system or other engineered safety features. The applicant will perform further evaluations of the effects of protective system failures. We will continue our review of this design area during the operating license review.

15.5 Hydrogen Control (Section 5.4)

The Committee noted that the applicant has proposed a purge system to control the potential buildup of hydrogen in the containment following a loss-of-coolant accident. We have concluded from our studies to date that a purge-type control technique may not be considered acceptable as the primary

means for hydrogen disposal. We have informed the applicant that an alternate technique should be developed. We will pursue this design area with the applicant during the construction phase.

15.6 Integrity of Vital Post-Accident Systems

The Committe indicated that potential problems in vital cooling systems may arise during the post-accident period as a result of severe environments and degraded coolant. The applicant has stated he will assure operability of vital systems by appropriate design and testing. We will evaluate the adequacy of the applicant's efforts during the operating license review.

15.7 Testability of Relay Circuitry

The Committee noted that the applicant has proposed a partial continuity check for the operability of the relay circuitry for engineered safety features. We have asked the applicant to develop a more complete testing technique. We will evaluate the applicant's efforts along these lines during plant construction.

15.8 Effects of Earthquakes on Structures and Components

The Committee feels that experimental verification of the anticipated behavior of important structures and components during large earthquakes and appropriate instrumentation to observe behavior of these items during smaller earthquakes should be considered. This information would provide guidance for continued reactor operation in the event of an earthquake. The applicant has indicated that he will consider these aspects further. We will pursue this design area during plant construction.

The report of the ACRS concluded "...The Committe believes that the items mentioned can be resolved during construction, and that, if due consideration is given to the foregoing, Nuclear Unit 2 proposed for the Diablo Canyon Site can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public."

16.0 COMMON DEFENSE AND SECURITY

The applicant reflects that the activities to be conducted would be within the jurisdiction of the United States and that all of the directors and principal officers of the applicant are American citizens. We find nothing in the application or otherwise to suggest that the applicant is owned, controlled or dominated by an alien, a foreign corporation or a foreign Government. The activities to be conducted do not involve any restricted data, but the applicant has agreed to safeguard any such data which might become involved in accordance with the regulations. The applicant will obtain fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material from military purposed is involved. For these reasons and in the absence of any information to the contrary, we have found that the activities to be performed will not be inimical to the common defense and security.

17.0 CONCLUSIONS

Based on the proposed design of the Pacific Gas and Electric Company's Diablo Canyon Unit 2 facility, on the criteria, principles and design arrangements for systems and components thus far described that include all of the important safety items, on the calculated potential consequences of routine

and accidental releases of radioactive materials to the environs, on the scope of the development program which will be conducted, and on the technical competence of the applicant and the principal contractors, we have concluded that, in accordance with the provisions of paragraph 50.35(a), 10 CFR Part 50 and paragraph 2.104(b) 10 CFR 2:

1. The applicant has described the proposed design of the facility, including, the principal architectural and engineering criteria for the design, and has identified the major features or components for the protection of the health and safety of the public;
2. Such further technical or design information as may be required to complete the safety analysis and which can reasonably be left for later consideration, will be supplied in the final safety analysis reports;
3. Safety features or components, which require research and development have been described by the applicant, and the applicant has identified, and there will be conducted, research and development programs reasonably designed to resolve any safety questions associated with such features or components;
4. On the basis of the foregoing, there is reasonable assurance that (i) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility and (ii) taking into consideration the site criteria contained in 10 CFR Part 100, the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public;

5. The applicant is technically qualified to design and construct the proposed facility; and
6. The issuance of permits for the construction of the facility will not be inimical to the common defense and security or to the health and safety of the public.

APPENDIX A

CHRONOLOGY OF THE PG&E APPLICATION REVIEW

DIABLO CANYON UNIT 2

ITEM	DATE	COMMENTS
Application for Construction Permit filed for Unit 2	7-1-68	PSAR Volume 1, 2, and 3 (Docket No. 50-323)
Initial meeting with applicant	8-13-68	Introductory session
Second meeting with applicant	10-22-68	Technical discussion
Third meeting with applicant	11-4-68	Technical discussion
Staff question list #1	12-11-68	Leter, PAMorris to RHPeterson
Amendment #1	5-12-69	Partial response to question list #1
Amendment #2	5-28-69	Partial response to question list #1
Amendment #3	6-23-69	Partial response to question list #1
Fourth meeting with applicant	7-30-69 7-31-69	Technical Discussion
Amendment #5	9-08-69	Clarification of responses to question list #1 plus additional information
Amendment #6	9-25-69	Supplementary meteorological data
ACRS Subcommittee Meeting	10-01-69	
ACRS Full Committee Meeting	10-10-69	