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SEP 20 1967

Docket No. 50-275

Mr. Nunzio J. Palladino
Chairman, Advisory Committee
on Reactor Safeguards
U. S. Atomic Energy Commission
Washington, D. C.

Dear Mr. Palladino:

Twenty copies of a report prepared by the Division of Reactor Licensing are transmitted for the review of the Committee. It is our Report No. 1 to the Committee in regard to Pacific Gas and Electric Company's Diablo Canyon Nuclear Plant.

Sincerely yours,

(Signed) Marvin M. Mann for
Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
ACRS Report (20 cys)

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September 20, 1967

U. S. ATOMIC ENERGY COMMISSION

DIVISION OF REACTOR LICENSING

REPORT TO THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

IN THE MATTER OF

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR PLANT

REPORT NO. 1

Note by the Director, Division of Reactor Licensing

The attached report has been prepared by the Division of Reactor Licensing for use by the Advisory Committee on Reactor Safeguards at its October 1967 meeting.

ABSTRACT

The Pacific Gas and Electric Company has proposed to build a PWR (Westinghouse) reactor plant at its Diablo Canyon site located adjacent to the Pacific Ocean in central California. The proposed plant will be operated at an average power density 18% above the Indian Point No. 2 facility which is, in other respects, similar to the Diablo Canyon plant.

This is the first report to the Committee concerning our safety review of this reactor project. In this report we have presented our evaluation of certain unique features that are of significance in our safety review. In this report we have included our evaluation of the site, the proposed seismic design bases, and the proposed core design in terms of the higher power density mode of operation. In addition, we have discussed, on a preliminary basis, certain aspects of the instrumentation, control, and auxiliary power systems.

On the basis of our evaluation of previous PWR's and the Pacific Gas and Electric Company's proposal of Diablo Canyon, we have made the following findings:

- the proposed site is suitable for the construction of the proposed facility
- the proposed seismic design bases for the containment and Class I piping are acceptable
- the proposed emergency power system is acceptable.

We plan to present our final safety evaluation to the Committee at its December 1967 meeting.

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1.0 INTRODUCTION

The Pacific Gas and Electric Company has submitted an application, dated January 16, 1967, for a construction permit and facility license for its Diablo Canyon Nuclear Power Plant. The facility will be located on a site adjacent to the Pacific Ocean in central California. The applicant has proposed to build a PWR four-loop plant similar to Indian Point 2 but with an 18% increase in core average power density. The reactor plant will be furnished by Westinghouse Electric Company. The reactor containment structure, which encloses the reactor and the steam generators, consists of a steel lined concrete shell in the form of a reinforced concrete vertical cylinder with a flat base and a hemispherical dome. The Diablo Canyon containment configuration, the free volume of the containment (2.6×10^6 ft^3), and the design pressure (47 psig) are the same as for the Indian Point 2 plant.

This report will provide the Committee with our preliminary evaluation of the site, the seismic design, and core physics, thermal and hydraulic design. The special features of each can be characterized as: (1) for the site, we must give special consideration to the seismic aspects; (2) for the seismic design, we must be convinced that the proposed design criteria will assure an adequate design with a high degree of safety under various operating conditions and accidents; and (3) for the core design, we must consider the safety aspects related to the proposed higher power density mode of operation.

In addition, we have included a section on instrumentation, control, and power systems. Our evaluation of the instrumentation and control system is not complete. However, our evaluation of the auxiliary electric power system has been completed and is included in this report.

It is our intention to complete our review of this application in these areas (as well as others) before the Committee's December 1967 meeting.

2.0 SITE CHARACTERISTICS

The Diablo Canyon site contains approximately 800 acres and is located adjacent to the Pacific Ocean and Diablo Canyon Creek in San Luis Obispo County, California. It is approximately 10 miles from the nearest boundary of San Luis Obispo (1965 population - 25,750). PG&E has leased the site land for 99 years with an option to renew for an additional 99 years.

2.1 Population Distribution

There are no communities within 6 miles of the site. The nearest residence is approximately 1-3/4 miles from the site. The 1960 cumulative population distribution as a function of distance from the proposed site and the projected 1980 population distribution are presented in Table 2.1 which follows.

TABLE 2.1

CUMULATIVE POPULATION DISTRIBUTION

AROUND THE DIABLO CANYON SITE AS A FUNCTION OF DISTANCE

Distance	1960	1980
1	0	0
2	4	4
3	6	6
4	10	24
5	12	76
10	1,572	6,902
20	49,202	118,362
30	87,182	208,862
40	122,072	287,662
50	148,592	344,262
60	157,982	370,992

The population data shows that the area is very sparsely populated out to a distance of approximately 6 miles. The land surrounding the site for some years will be for low density housing and recreational development.

The minimum exclusion distance is defined in 10 CFR 100 for the Diablo Canyon site as 0.5 miles. The population center distance is 10 miles, which is the distance from the site to the nearest boundary of San Luis Obispo. Since the population density out to 10 miles is quite low and the 10 CFR 100 guidelines state that the distance to the nearest boundary of the closest population center should be at least 1-1/3 times the calculated low population distance, we take 7.5 miles as the low population distance for calculational purposes. Our preliminary calculations indicate that for the 10 CFR 100 postulated releases, exposure criteria can be satisfied provided that some credit for iodine removal (factor of 3 for the 2-hour dose at 1/2 mile) and that the low population zone is adequate with regard to the 10 CFR 100 guidelines.

2.2 Meteorology

Conservative diffusion climatology for the Diablo Canyon site has been used by the applicant in lieu of on-site meteorological data. ESSA has reviewed the information on meteorology in the PSAR and concluded in their comments which were forwarded to the ACRS, that the meteorological assumptions described by the applicant in the PSAR are conservative. To verify the meteorological assumptions used in the PSAR, the applicant proposes an ambitious meteorological program prior to plant operation which includes meteorological measurements on a 250-foot tower near the plant location and on a 100-foot tower at the top of the 914-foot hill on the site. Tracer diffusion studies using flourescent

particles and smoke will also be performed. We feel that the proposed meteorological program is quite adequate to provide a firm basis for the development of a gaseous radioactive release limit and to confirm the conservatism of diffusion parameters used for the evaluation of the consequences of accidents.

2.3 Hydrology

The hydrology of the site does not appear to present any potential problems for this site as there is little or no probability of contamination of domestic water supplies (the nearest open reservoir is thirteen miles northeast of the site and surface drainage is expected to be toward the ocean) and the Diablo Canyon Creek with a drainage basin of four square miles is incapable of a flood that could endanger the site.

2.4 Geology

The site has been extensively trenched to a depth of from 10 to 40 feet by the applicant to identify the geologic characteristics of the site (see Figure II-A-1 in Supplement 3). All of the Class I structures will be founded upon bedrock, which is made up of marine shales, sandstones, and fine grained tuffaceous sediments. We have been told informally by the U.S.G.S. that the bedrock is quite adequate to be used for the foundation of the facility. The formal report of our consultants from the U.S.G.S. will be forwarded to the ACRS as soon as it is available.

Minor inactive faults have been traced through the site, including a fault or slip zone which runs under the proposed location of the reactor containment. The strata covering this fault is undisturbed, indicating that the last movement of this fault occurred at least more than 100,000 years ago, and probably

more than a million years ago. The applicant's consultants feel that there is little or no likelihood of movement along this fault. Our geologic consultants have informally told us that they agree with this conclusion.

We feel that the geology of the site should present no unusual engineering problems for the construction of this nuclear facility.

2.5 Oceanography

Condenser cooling water will be provided by the Pacific Ocean. PG&E reports that the liquid radioactive wastes will be discharged with the condenser cooling water at or below the 10 CFR 20 limits. The effects of reconcentration in aquatic biota will be considered by PG&E in its monitoring program. Details of the monitoring of the aquatic environs proposed by the applicant are discussed in a later section of this report.

The applicant has analyzed the potential for flooding of the site by tsunamis. It should be noted that all of the Class I structures and equipment are located 80 or more feet above MSL (Mean Sea Level) except the intake structure. The top of this structure will be 20 feet above MLLW (Mean Low Low Water). The peak tsunami wave height, which includes peak storm and high tide and run-up is approximately 18 feet above MLLW providing a minimum freeboard for any Class I structure of 2 feet. The maximum draw-down due to tsunami and low tide is 9 feet below MLLW. We have been told informally by our consultants in the USC&GS that they feel that in order to provide adequate tsunami protection, the minimum protection level should be approximately 30 feet above MLLW. The applicant has been informed of this and has orally indicated that they will comply with our consultant's recommendations.

2.6 Seismology

The applicant has studied the seismic history of the Diablo Canyon area and has determined the maximum earthquakes relative to the faults in the general area. On the basis of this investigation, the applicant concludes that there are four possible types of earthquakes that would result in maximum accelerations at the site. These will establish the design basis for the Diablo Canyon plant. The maximum ground accelerations considered by the applicant were:

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

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

Earthquake C: A 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 0.05g at the site.

Earthquake D: After-shock with a magnitude 6-3/4 at the site associated with earthquake A, above, which results in a ground acceleration of 0.20g at the site.

We have been informed by our seismic consultants of the USC&GS that they feel that a design earthquake with a horizontal ground accleration of 0.20g and that a maximum credible earthquake, or safe shut down condition, with a horizontal ground accelration of 0.40g should be used for this site. The applicant reports that a strong-motion seismograph would be installed in the

facility prior to plant operation. The seismic design aspects associated with the Diablo Canyon plant are discussed in a later section of this report.

2.7 Environmental Radioactivity Monitoring

The applicant has stated that, to establish background radioactivity data, an environmental monitoring program will be initiated at least two years prior to plant operation. They propose to monitor airborne gamma activity, air particulate activity, bovine thyroid, milk, leafy vegetables, and aquatic flora and fauna. We feel that the program proposed by the applicant will provide a firm basis upon which the post-operational environmental radioactivity monitoring program can be developed. Comments of the Fish and Wildlife Service, Department of the Interior, have been forwarded to the ACRS and the applicant.

3.0 SEISMIC DESIGN EVALUATION

3.1 General

This section of the report is based on the information included in the Preliminary Safety Analysis Report (PSAR) and Supplements 1 through 3, a preliminary report from our consultants, Drs. Newmark and Hall, and discussions with PG&E and Westinghouse personnel.

In this section we will present our evaluation of the adequacy of the proposed criteria for the response of Class I structures, systems, and components to seismic forces in combination with other applicable loads.

The simultaneity of an earthquake with a loss-of-coolant accident has been accepted by PG&E and other applicants for the design of containment structures.

We believe that this concept should be extended to all Class I structures, systems, and components including the containment structures, the reactor coolant system, the reactor vessel, the reactor vessel internals, the emergency core cooling system, other engineered safety features, and vital support structures and members for such systems and components. The applicant has agreed to follow this approach and has advised us that with certain modifications, the plant could be made to withstand simultaneous earthquake and blowdown forces. These modifications include changes to the support saddle for the reactor vessel (under the nozzles) and to the reactor internals. Westinghouse has indicated, however, that with the present reactor vessel and reactor core designs, the reactor internals could not be modified without permitting the combined loading stresses to exceed ASME Code Section III allowable values for stresses. If the modifications were made to conform to code allowable stresses, the thermal and hydraulic performance of the proposed high power density core would necessitate derating of plant power. In any case, we intend to be assured that the plant is designed to accommodate the effects of a simultaneous earthquake and coolant loss accident. We will consider limits that may not conform with code limits under certain conditions, provided the applicant can justify the basis with a high degree of confidence.

3.2 Earthquake Magnitudes

The earthquake loadings for design purposes will be based on two postulated earthquakes. Earthquakes B and D (Section 2.6) were determined by the applicant to be controlling for plant design purposes at this proposed site. Earthquake D has been assumed to be the "close-by" earthquake since its reponse spectrum

(based on the Golden Gate 1957 earthquake) is maximized in the short period range and decays rapidly as the period increases. The period range where Earthquake D maximizes is for periods less than 0.2 to 0.3 seconds. Maximum ground acceleration is predicted to be 0.20g.

For longer periods, maximum accelerations occur for Earthquake B. The response spectrum corresponds to the 1952 Taft earthquake. This may be characterized as the "far-away" earthquake. Maximum ground acceleration for this earthquake is normalized to 0.15g for design purposes as compared to the original estimate by PG&E of 0.12g.

For design purposes, the applicant proposes to use both an envelope of the B and D response spectra as well as B and D spectra separately. This would encompass Earthquake B using the Taft response spectra with a horizontal acceleration of 0.15g and Earthquake D using the Golden Gate response spectra with a horizontal acceleration of 0.20g. The applicant further reports that the design will be evaluated in terms of "safe-shut" down for which the maximum accelerations will be increased by a factor of two.

Vertical acceleration values in all cases will be taken as two-thirds of the corresponding maximum horizontal ground acceleration and the effects of horizontal and vertical earthquake loadings will be combined, and considered to act simultaneously. This is in agreement with Dr. Newmark's proposed specification in the "Seismic Design Crtieria for Nuclear Power Plants" (May 1967).

3.3 Response Spectra and Damping

The applicant has specified response spectra for the assumed earthquakes along with an envelope of the spectra for the no-loss-of-function condition.

We concur with the opinion of our consultants (Drs. Newmark and Hall) that these spectra are acceptable provided that both earthquakes (B and D) are used and the maximum response in either must be considered to apply to the design for safe shutdown of single-degree-of-freedom elements. In the opinion of Dr. Newmark, this is acceptable since Earthquake B gives response values for low and intermediate frequencies that lie above the response spectrum values from TID-7024 when normalized to an acceleration of 0.40g. In effect, this earthquake corresponds to a 0.40g earthquake for low and intermediate frequencies. For a safe shutdown of a multi-degree-of-freedom system, Dr. Newmark has indicated that the envelope spectrum for both earthquakes should be used. This envelope spectrum is consistent with the El Centro type response spectrum for a maximum ground acceleration of 0.40g. We intend to discuss these aspects with PG&E and will be prepared to provide the Committee with the results at the October meeting.

We have reviewed the damping values to be used in the design and concur with the selected values. They compare favorably with the values listed in Table 1 of the "Seismic Design Crtieria for Nuclear Power Plants" by N. M. Newmark and W. J. Hall (May 1967).

3.4 Load Combinations and Stress Limits

The following is a summary of the applicant's preliminary position in regard to the seismic design. This information has neither been provided formally nor with any supporting analysis to permit an evaluation at this time, however, it does provide a basis for current understanding.

3.4.1 Design Earthquake

(a) The seismic design of Class I equipment will be based on the accelerations and resultant loadings from the "B" and "D" earthquake response spectra as presented in the Third Supplement to the PSAR. These response spectra will be treated as separate loading conditions. Stress levels in code designed equipment will be within code limits. Vital components that do not fall within code jurisdictions, such as the reactor internals, will use stress levels of Section III of the ASME Boiler and Pressure Vessel Code as a guide.

(b) The design of Class I equipment will also be evaluated for a response curve representing the envelope spectrum of earthquakes "B" and "D". Under this condition, stress levels in code-designed equipment will be within code-allowable limits. Stress levels in the reactor internals may exceed the allowable stress levels of Section III which is used as a guide. Under such conditions, increased damping coefficients may be used for the design of core internals where sufficient evidence and justification are available.

3.4.2 Maximum or No-Loss-of-Function Earthquake

(a) Analysis of Class I equipment for the no-loss-of-function earthquake will be based on response curves equal to twice the envelope of the combined "B" and "D" spectra. Loadings from the above spectra will be combined with the functional loads.

(b) Where appropriate, loadings from the above spectra will also be combined with functional loads and blowdown forces resulting from a major loss-of-coolant accident. The criterion for this combination of loads will be, as before, the ability to shut down the reactor and establish emergency core cooling. For the case of piping and pressure vessels, evaluated stresses will remain within the limit curves of WCAP-5890, submitted as Appendix A of Supplement 1.

We are currently reviewing the applicant's criteria for load combinations, but believe that more definitive information is necessary before we can determine their acceptability in all respects. This information would include specific stress limits to be used for the Class I items, including the reactor internals, for the design and maximum earthquake cases. If higher damping coefficients are to be used for the design of the vessel internals, we would need more details of the applicant's analysis. We believe this information can be made available prior to the December ACRS meeting. PG&E has agreed to provide us with sample calculations on representative piping and reactor internals to clarify their design criteria.

We concur generally with the position of Dr. Newmark that revised damping factors can be considered without compromising safety, provided this is done as a function of stress and deformation level. We are awaiting PG&E's submission of supporting information in this area. We intend to have Dr. Newmark or his representative available at the October 1967 Committee meeting for further discussion of this matter.

We believe that the design stress criteria and the load factor expressions to be employed in the design of the containment are reasonable. This will be discussed further in a later section. However, our review of the design approach for the stress criteria for other Class I components, as presented in the WCAP-5890, "Ultimate Strength Criteria to Ensure No Loss of Function of Piping and Vessels Under Earthquake Loading," is continuing. While we accept the application of limit analysis principles, we need to establish the validity and conservatism of the limit curves presented in WCAP-5890 as applied to the Diablo Canyon plant design.

3.5 Dynamic Analysis

The applicant has reported that the dynamic analysis associated with the Class I components and structures will be based on the modal participation factor method. This method is outlined in "Design of Multistory Reinforced Concrete Buildings for Earthquake Motions" by John A. Blume, Nathan M. Newmark and Leo H. Corning. Dr. Newmark, in his draft report on the Diablo Canyon plant, dated September 7, 1967, concurs with the use of this method for multi-degree-of-freedom systems. For single degree of freedom systems, the applicant proposes, and we agree, to use the natural mode for vibration in the analysis.

We understand that the modal analysis will be carried out using either the smoothed spectra, directly or by using a time history of ground motion, employing earthquake records with scaled amplitude values which the applicant claims will give essentially the same smoothed spectra. Dr. Newmark concurs with this approach provided that the time history input yields the same response

spectra without any major deviations below those smoothed response spectrum values presented in the PSAR. We are in the process of clarifying this point with our consultants and will be prepared to discuss it further at the October meeting.

As a further point on the dynamic analysis, it is our understanding that the design of Class I components, particularly for the safe shutdown conditions will be made for the envelope of the combined spectra of the two earthquakes for the appropriate damping level. We agree with this approach.

3.6 Design of Structures, Systems and Components

3.6.1 Containment Structure

The reactor containment structure consists of a steel lined concrete shell in the form of a reinforced concrete vertical cylinder with a flat base and a hemispherical dome. The applicant reports that the concrete reinforcing steel pattern consists of bars oriented at 30° from the vertical in such a manner that the pattern does not require termination of any bars in the dome. These bars are designed to carry both the lateral shear as well as vertical tensile forces. Hoop reinforcing is provided for the cylindrical portion of the structure. For radial shear reinforcing, the applicant proposes to use a system of vertical wide flange beams spaced four feet on centers. The beams are attached by hinge connection to the base slab at the lower end and are terminated about 20 ft above the top of the base slab. The beams provide resistance to the moments and shears created by the discontinuity at the base and provide a gradual transition of load carrying elements between the base and the cylinder wall.

The proposed method for carrying the radial shear is novel. Dr. Newmark has indicated that this application is acceptable. We have asked PG&E to give careful attention to the detail at the base of the I-section where it is keyed into the foundation, to insure that no distress can occur in either the liner or the diagonal reinforcing bars through any rotation that might occur at this point under earthquake loadings and/or accident loadings.

The factored load combinations and design stress criteria for the containment are acceptable to us and to Dr. Newmark. The applicant has stated that no steel reinforcement will experience average stress beyond the yield point at the factored load conditions. Also, the statement is made that the liner will be designed so that stresses will not exceed the yield point at the factored load conditions. We interpret these statements to mean that the average stress in the reinforcement and liners will not exceed yield and that the deformations will be limited to that of general yielding under the maximum earthquake loading conditions. We are in the process of verifying this point with the applicant and will be prepared to discuss this matter at the October meeting.

Based upon our review and that of Dr. Newmark's, it is our opinion that the applicant's seismic design criteria for the containment structure are acceptable. We note that there are a few areas yet to be clarified. These are also identified in Dr. Newmark's draft report which has been provided to the Committee. We intend to resolve these matters by the Committee's October meeting.

3.6.2 Penetrations

The applicant reports that for large penetrations, the diagonal rebars will be welded directly to a heavy structural steel ring through use of Cadweld sleeves. We believe that this approach is satisfactory. Dr. Newmark has reported that the applicant indicated that the stress concentration in the vicinity of the opening is to be considered in the analysis. Although he indicates that this approach may be satisfactory, he believes, and we agree, that the penetrations design should take account of any secondary effects arising from local bending, and thermal effects to insure that the penetration-door detail behaves satisfactorily. Partial proof of the integrity of the penetration will be provided by the measurement program to be made concurrently with the proof testing of the containment vessel. Based on a recommendation from our consultants, we are asking the applicant to calculate the penetration deformation prior to the proof testing to provide evidence that the design does indeed meet the criteria set forth for both the large and small penetrations. We will report on this at the October meeting.

3.6.3 Intake Structure

The intake structure will be designed as a Class I structure, with due regard for predicted tsunami water heights. Although the applicant indicates that some protection will be provided against the possibility of rock masses from the cliff falling onto, or into, the pump house, our consultants recommend that careful attention be given to any possible impairment of the controls or the pumping system through any possible rock falls or slides. We intend to review this, as well as the tsunami height problem previously discussed, with the applicant.

3.6.4 Reactor Internals

The design of the reactor internals has been reviewed with the applicant. They are to be designed to withstand the combined maximum earthquake spectrum concurrent with blowdown in such a manner that while moderate yielding may occur it would not impair core cooling capability. This matter is under detailed study and further information will be provided for our evaluation. We intend to discuss the results of this evaluation in a subsequent report to the Committee.

3.6.5 Reactor Coolant System

Class I piping will be designed to the USA S.I.B31.1 Code for pressure piping which includes consideration of internal pressure, dead load, and other appropriate loads such as thermal expansion. The applicant indicated, that earthquake effects will be considered with these loadings and further elaboration of this point is given in Appendix A of Supplement 1 (WCAP-5890).

The applicant has indicated that there may be regions of local bending where the stresses in the piping could be equivalent to 120% of yield. However, the design bases for the piping system include the requirement that these local deformations will not cause a loss of service capability. Dr. Newmark has indicated, and we agree, that the deformations should be limited so that a loss of function would not occur. PG&E has agreed that this matter would receive full consideration during design. In the interim, PG&E has indicated it would provide us with the results of a sample calculation to better illustrate its design approach.

3.7 Conclusions on Seismic Design

On the basis of the information presented and outlined in this section, we believe that the proposed design criteria for the containment and the Class I piping will provide an adequate margin of safety to withstand seismic loads.

We believe, however, that more definitive information on the criteria for all Class I components and structures other than the containment and piping, and in particular, the reactor vessel internals must be provided to complete our review.

4.0 CORE THERMAL, HYDRAULIC AND PHYSICS DESIGN

4.1 Design Comparison

The proposed plant is reported by the applicant to be similar to Indian Point II with certain design differences. These differences are apparent in certain equipment types and sizing due to the increased average power density and rating for the Diablo Canyon plant. The average power density is approximately 18% greater than Indian Point II. The principal differences between the two designs in terms of the power uprating are listed in Table 4.2.

In addition to these differences, the applicant reported in Supplement 3 that the moderator temperature coefficient of reactivity will be negative at all times throughout all operating cycles. Burnable poison will be placed in certain unused rod cluster control tubes in selected assemblies within the core. Previous designs on other similar PWR plants included the potential for a positive moderator coefficient of reactivity.

Further, it was reported in Supplement 1 that part length control rods (eight such assemblies) are to be provided in the reactor in addition to the normal control rods. These rods will be used for power distribution shaping and to control potential axial xenon oscillations. This design feature is new and therefore has not been evaluated on previous plants. We plan to evaluate the bases and significance of the foregoing design differences in terms of plant safety for both normal operation including anticipated transients as well as the full spectrum of potential accident conditions. The overall aspects of the reactor design in the areas of core thermal hydraulic and physics design are treated on a preliminary basis in this section of this report. Our final position on these matters will be presented in a subsequent report to the Committee.

4.2 Thermal and Hydraulic Design

The Diablo Canyon plant design proposes the highest power density PWR application received to date. This plant follows the general trend apparent over the past few years of increasing average and peak heat fluxes (see Figures 4.2.1 and 4.2.2). A comparison of the significant thermal and hydraulic characteristics is presented in Table 4.2.

We note that although the average and the peak heat fluxes have increased over the past few years, the minimum DNB (Departure from Nucleate Boiling) ratio has remained about constant. Using the W-3 correlation, most recent Westinghouse reactors have a DNB ratio between 1.81 and 1.90. It has been possible to maintain a "constant" DNB ratio in spite of increased heat fluxes by some optimization of the inlet enthalpy, but principally, by using lower peak to average factors.

An example of this is given by comparing the thermal-hydraulic characteristics of Diablo Canyon to Indian Point II (see Table 4.2). These two reactors are nearly identical except that the inlet temperature of Diablo Canyon is about 11°F lower. The lower inlet temperature results in a lower enthalpy throughout the core which in turn produces, according to the W-3 correlation, a higher burnout heat flux. The combination of lower enthalpy and higher heat flux of Diablo Canyon results in maintaining the DNB ratio within 1.81.

This consequence results in the reduction of the ratio of core peak to average heat flux values. Peak to average values of heat fluxes for applications received from 1963 to mid-1966 ranged from 3.16 to 3.42, but generally were at about 3.25. These values include ratios for plants like Connecticut Yankee, GINNA, Indian Point #2, Turkey Point #3, H. B. Robinson, and Pt. Beach #1. From mid-1966 to the present time, Westinghouse reactors have had peak to average values of about 2.81. These include applications such as Diablo Canyon, Surry, Indian Point

#3, and Kewaunee. According to the applicant, the peak to average factors have been reduced due to improved physics calculations and by control of the axial power distribution which is made possible by use of the proposed part length control rods. The improved physics calculations represent more recent analytical methods. The part length rods can be used to reduce xenon oscillations and to flatten the axial power distribution. It is our understanding that the application of part length rods will be used on other similar Westinghouse PWR reactor plants prior to the Diablo Canyon plant.

Maintaining a particular minimum DNBR is not a complete description of the thermal characteristics of a reactor. For example, if a core could be designed with a peak to average ratio of 1.0, and then operated at a DNBR of 1.81, this "flat power" plant could affect safety to a greater extent than the actual Diablo Canyon design in spite of equal DNBR's. For example, an error in the thermal analysis of Diablo Canyon might put a few rods at DNB during an accident, while every fuel rod in the entire "flat power" core would be at DNB for the same accident. One is therefore also interested in the numerical distribution of rods at various DNB ratios as well as the minimum DNB ratio.

Our evaluation of the adequacy of the Diablo Canyon thermal design will be based upon consideration of all potential thermal limits. These limits will include:

(a) burnout heat flux distribution

(b) fuel centerline temperatures

(c) transient effects (e.g., loss-of-flow accident)

(d) hydraulic stability

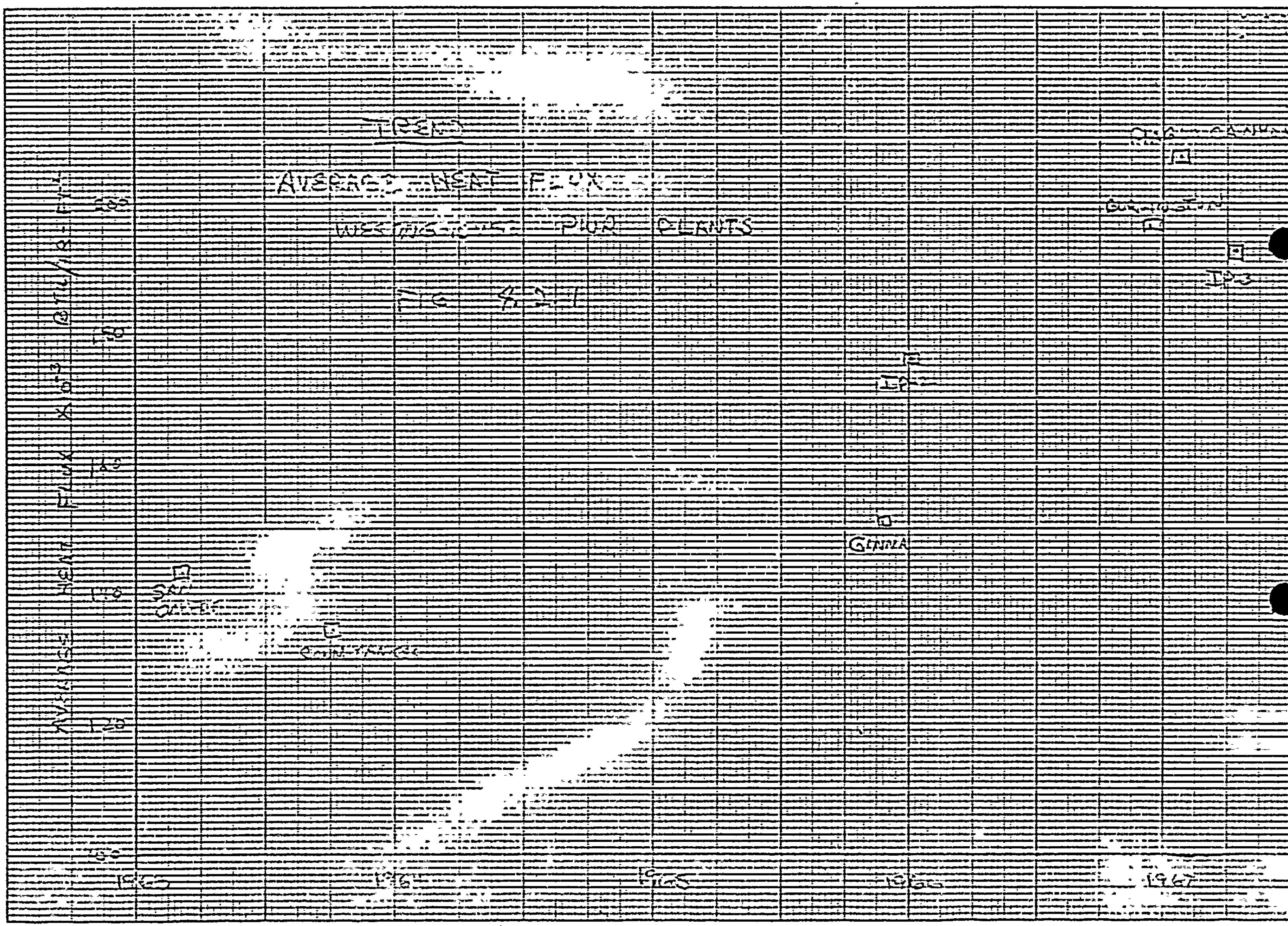
(e) core heatup after a loss of coolant.

TABLE 4.2

COMPARISON OF THERMAL AND HYDRAULIC PARAMETERS

ITEM	Diablo Canyon	Indian Point II	Indian Point III	GINNA (Brookwood)	San Onofre
Maximum Specific Power, Kw/ft	18.9	18.5	17.6	16.7	15.0
Maximum Heat Flux, $\frac{BTU}{HR\text{-}FT^2}$	583,000	570,000	543,000	517,000	463,000
Average Heat Flux, $\frac{BTU}{HR\text{-}FT^2}$	207,000	175,600	193,000	151,000	143,000
Average Mass Velocity, $\frac{lb}{HR\text{-}FT^2}$	2.54×10^6	2.56×10^6	2.53×10^6	2.43×10^6	2.02×10^6
Nominal Inlet Temperature, °F	539	543	549.7	556	553
Core T, °F (Average)	68.6	57	63.2	54	49
DNBR at Nominal Conditions	1.81	1.81	1.81	1.90	2.07
F_q	2.82	3.25	2.82	3.41	3.23
$F_{\Delta H}$	1.70	1.88	1.70	1.88	1.88
Date of Receipt of PSAR	1/18/67	1/6/66	4/26/67	11/2/65	2/4/63

OFFICIAL USE ONLY


TREND
AVERAGE HEAT FLUX
PWR PLANTS
GINNA
1967

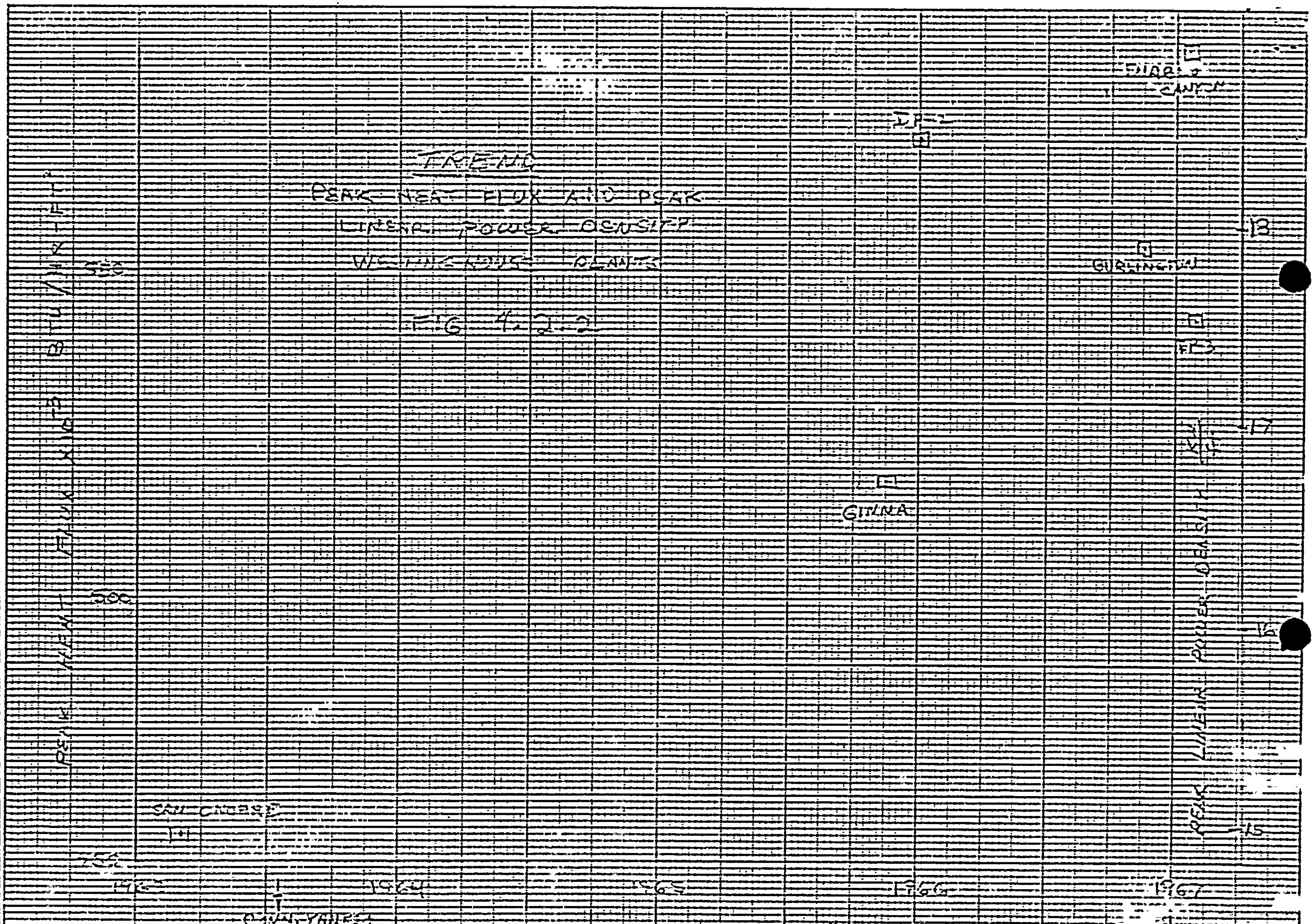


FIG 4.2.2

437,000

We are evaluating the significance of the increased power density in these areas. Our final position will be discussed in a subsequent report to the Committee.

4.3 Core Physics

Part length control rods are to be provided in the reactor in addition to the normal control rods. There will be eight such assemblies, with control material only in the bottom three feet. The function of these rods is to shape the axial power distribution and to permit control of potential axial time dependent oscillations of power distribution caused by differential xenon concentrations. The applicant has furnished analysis which illustrates in principle that the part length rods can perform their intended function. This design feature for Westinghouse PWR's has not been analyzed by us before and therefore will require continued review by us as information becomes available from the applicant. Factors to be included are the following:

(a) selection of the optimum core lattice locations for the part length rods.

(b) definition of the administrative control limits on part length rod travel as a function of normal rod insertion.

(c) definition, as a function of normal rod position, of the relationship between upper and lower half core power levels to be maintained with the part length rods, as a function of power level.

(d) determination of timing of part length rod position adjustments.

(e) analysis of possible effects of individual part length rod operations, either by accident or purpose.

(f) evaluation of effects of the part length rods on the spectrum of accidents analyzed for the reactor.

(g) detailed analysis of the power distribution under various transient conditions.

(h) the mechanical design of the rods, including cooling provisions, etc.

(i) experimental and analytic evidence of the ability to reliably and accurately determine the presence of core power imbalances.

With regard to the latter point, the applicant proposes to rely primarily upon external neutron detectors. These will consist of four long ion chambers, each divided to measure flux in upper and lower halves of the core. The problem of evaluation of the external detector concept is really separate from that of use of part length rods, because it is necessary to have indication of radial and azimuthal power imbalances as well as axial. It is currently not known whether external detectors can provide sufficient information for these functions. The applicant reports that tests with long chambers will be performed in the SENA reactor to determine their effectiveness in monitoring core power distributions. In addition, we have initiated studies at the Brookhaven National Laboratory to evaluate the problem. A final decision on the adequacy of external core detectors awaits development of further information. The problem is not unique in the Diablo Canyon plant design, but could be of greater significance in this plant because of the proposed higher power density. We understand that this matter will be the subject of an appropriate test program and based upon the applicant's response to our questions, such a program is being developed.

5.0 INSTRUMENTATION, CONTROL, AND POWER SYSTEMS

5.1 Instrumentation and Control

Our evaluation of the instrumentation and control systems is not complete. The only differences the applicant anticipates between the Diablo Canyon instrumentation and control systems and those of H. B. Robinson and Point Beach plants result from the different number of primary loops. Our evaluation will be a part of the continuing evaluation of the Westinghouse designs for instrumentation and

control systems. We are investigating the ability of equipment in engineered safety feature systems to function in the accident environment. This includes instrumentation and electrical equipment which is relied upon to function under accident conditions inside containment. We intend to pursue areas which in the past have been evaluated only on the basis of the applicant's criteria. Since criteria require specific interpretation, it is desirable to determine how the criteria are to be implemented. Much of the design, however, still appears not to be firm. The protection systems, therefore, may have to be evaluated largely on the basis of criteria alone. We will determine how the applicant plans for the specific implementation of the criteria.

5.2 Auxiliary Electric Power

The engineered safety feature loads are connected to three 4160 volt vital busses. When off-site power is not available, each bus is powered by a separate diesel generator. The redundancy of safety feature loads and the arrangement of loads on the busses is such that the required minimum of engineered safety features will be available after the loss of any one bus. The failure of one diesel to start will not prevent the required minimum of engineered safety features from operating from on-site power. The busses are operated in a split bus arrangement, and the generators are not paralleled.

Three station batteries are provided. The circuit breakers associated with each of the three vital busses receive their control power from a different battery. A single battery failure of d.c. circuit fault should disable no more than one of the three vital busses. A single failure in the d.c. circuit breaker control circuits should not prevent operation of the required minimum of engineered safety features from either off-site or on-site power.

Off-site power is supplied to the Diablo Canyon plant by two 230 kv transmission lines. The single startup transformer can be energized from either transmission line. Under accident conditions, the vital busses are connected to the startup transformer. We do not believe that the use of redundant startup transformers would materially improve the reliability of off-site power. A transformer's reliability tends to be high enough to prevent it from being the limiting item even when redundant lines and loads are provided. A motor operated link is provided to disconnect the main generator from its transformer, so that power can be brought into the plant from either of two 500 kv lines when the main generator is shut down. Although this source of off-site power would not be available immediately after an accident, it will improve the reliability of off-site power for post-accident cooling.

We believe that the on-site system can be designed to power minimum engineered safety feature loads under accident conditions with a simultaneous loss of off-site power and a single failure in the on-site electrical system. The proposed system, we believe, can be designed to provide reliable power to the station from off-site sources.

6.0 FUTURE REVIEW MATTERS

We are continuing our safety evaluation of the proposed Diablo Canyon reactor facility. Our efforts up to this time have been principally directed toward the acceptability of the site and the seismic design basis. Since the site is acceptable in our opinion, our evaluation will now be expanded to more fully evaluate other aspects of the design and proposed operation of the facility. We believe that our evaluation of following matters will be completed in time for the Committee's December 1967 meeting:

-- the design basis for the containment

-- the instrumentation and control systems

-- the core thermal, hydraulic, and physics design as related to the safety aspects of the high power density core

-- the adequacy of the proposed emergency core cooling system with respect to the properties of the high power density core

-- the design bases accident analyses to evaluate the adequacy of the plant's engineered safety features in terms of public health and safety.

7.0 CONCLUSIONS

On the basis of our evaluation concerning the particular review matters discussed in this report, the following conclusions are made:

-- the proposed site is suitable for construction of the proposed reactor facility in terms of those matters related to the site and environs (e.g., population, meteorology, hydrology, geology, oceanography, and seismology).

-- the proposed siesmic design bases for the containment and piping are acceptable.

-- the proposed emergency power system is acceptable.