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Docket No. 50-275

NOV 28 1967

Mr. Nunzio J. Palladino
Chairman, Advisory Committee
on Reactor Safeguards
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Palladino:

Twenty-four copies of a report prepared by the Division of Reactor Licensing are transmitted for the review by the Committee. The report is our second report on the proposed Pacific Gas & Electric Company's Diablo Canyon facility. The section on instrumentation is under preparation and will be transmitted as soon as possible.

Sincerely yours,

DRL RPT # 2
DTD - 11-28-67

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
ACRS Report (24 cys)

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NOV 28 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

DOCKET NO. 50-275

REPORT NO. 2

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 December 1967 meeting.

1.0 Introduction

Pacific Gas and Electric Company submitted an application dated January 16, 1967, for a construction permit for its proposed Diablo Canyon Nuclear Power Plant. A previous report to the ACRS dated September 20, 1967 was prepared which included our preliminary evaluation of the site, seismic design, core physics, and thermal-hydraulic design. This report presents the results of our evaluation of the proposed facility design in those areas where reservations were previously expressed as well as items not included in the previous report.

In certain areas the staff has not accepted the applicant's proposed design. We have informed the applicant of these areas and they are discussed in the following sections. It is our understanding that the applicant proposes to file an amendment prior to the December ACRS meeting date to formally document oral commitments.

2.0 Site Characteristics

In our first report to the Committee the only siting matter that was not resolved was the problem of suitable plant protection against potential tsunamis. The applicant has proposed that the use of a 20 foot tsunami (including peak storm and high tide) for protection design purposes was sufficiently conservative for this site and presented information in support of its view. This information was reviewed by our consultants in the USC&GS and ESSA. Based upon this review and a discussion with the applicant on November 21, 1967, our consultants have not changed their opinions and we believe with them that protection against flooding from a tsunami should be provided to an elevation of 30 feet above mean low low water. At the conclusion of this meeting, the applicant orally agreed to protect all

Class I structures to this elevation. As originally proposed, all Class I structures and equipment except the intake structure are located 80 or more feet above MSL. The top of the intake structure as designed would be 20 feet above MLLW (Mean Low Low Water) and to accomodate the added tsunami height a 10 foot wall will be built on top of the intake structure around the fire and auxiliary sea water pump motors (the pumps needed to maintain the nuclear facility in a safe shutdown condition) protecting them to a 30 foot level.

In the judgment of our consultants the maximum draw-down due to the tsunami could result in a lowering of the sea water level of approximately 25 feet below mean low low water. They further stated that the duration of the drawdown condition would be short, taking less than one hour for a complete cycle with only a few minutes at the maximum drawdown.

The applicant stated that the intake structure will be designed to provide a "wet well" of adequate capacity for assuring at all times a sufficient volume of water for operation of the auxiliary sea water pumps. This design concept provides for a weir type arrangement to trap water in the intake structure to a depth of about 12 feet. Under conditions of extreme drawdown, sufficient water would be trapped in the intake structure to permit operation of the auxiliary water pumps for approximately 30 minutes. This design will require shutdown of the main cooling water pumps when the drawdown exceeds a given elevation because these pumps also draw from the same source. To assure that the main cooling water pumps would be shut down the applicant has stated that they will receive warning of potential tsunami conditions through the ESSA alerting system. Upon receipt of the alert, the applicant stated that an observer, who will be in contact with

the control room, will be posted and when the water at the intake structure reaches a pre-set level the plant will be shut down. We and our consultants feel that with these design provisions the Diablo Canyon facility will be adequately protected against tsunamis. Selection of the level for shutdown and whether or not automatic protection is required are being deferred to the operating license review stage.

3.0 Seismic Design

Our previous report to the ACRS included a section on the seismic design criteria proposed for the Diablo Canyon facility. At that time our review of the containment design was complete except for a few outstanding items where clarification was requested from the applicant. The design criteria for other Class I structures was still under review at the time of our last report. Additional information, presented in Amendments 5 and 6, has been reviewed by both the staff and our consultants. Our positions for the containment structure and other Class I structures and components are discussed separately below. We expect that our consultants report will be available prior to the December ACRS meeting.

3.1 Containment Design

Factored loads for the design of the containment structure have been proposed which combine dead loads, pressure loads, temperature loads and earthquake loads (or wind load if greater than the earthquake load). The formulae for the three loading conditions are presented on page 5-10 of the PSAR. The containment will be designed such that the most restrictive loading combination for each particular region of the containment results in average stresses not greater than the yield point. The staff and our consultants concur in the design approach proposed by the applicant.

The reactor containment structure, consisting of a steel-lined, reinforced concrete, straight circular cylinder, with a hemispherical dome and a flat bottom, presents two new features; a helical reinforcing pattern in the concrete shell and a hinge at the base of the cylindrical wall. The concrete cylinder is reinforced with helical bars, inclined at an angle of 30° from the vertical. The wall reinforcing bars are continuous with the dome reinforcing. Additional hoop reinforcing is provided in the cylindrical wall. The continuity of the wall and dome reinforcing does not require termination and anchorage of any bar in the dome, and is an attractive feature of this reinforcing arrangement. Another advantage is the direct transmission of shears throughout the structure. The applicant presents a preliminary arrangement of the reinforcing pattern which will require further attention as outlined below. Detailed arrangement of the reinforcing bars including the location of the splices, the possible interferences between the bars, the erection sequence of the reinforcing, the arrangement of the reinforcing at special points such as openings, zones of discontinuities, groups of penetrations, have still to be worked out. We do not foresee any insurmountable problems in the preliminary design and recognize that alternate possibilities may be used if unexpected difficulties should arise during the final design stage.

The design at the base of the wall incorporates a system of vertical steel beams, spaced four feet on centers. The beams are hinged at their base and are 20 feet long. The base of the wall is divided into three concentric layers. The inner layer, approximately twelve inches thick, supports the liner. The intermediate layer, approximately 16 inches thick, contains the vertical steel beams anchored into concrete adjacent to them.

The exterior layer contains the helical and the hoop reinforcing bars. The two surfaces of contact between the three layers, and the steel beams will be treated with a bond-breaking substance, to ensure independent action of all elements. The purpose of this arrangement is to ensure transmission of the radial shears from the wall into the base. This is a new design and will require more studies and tests to clarify its behavior under all possible load combinations.

It is not clear how the stresses will be transmitted from the beams into the adjacent concrete slabs and vice versa. It is also not clear how the hinge action will be ensured across three layers of concrete. Finally, the rotation at the hinge may influence the behavior of the liner at this location in an unfavorable maneuver. However, if further studies disclose unexpected difficulties, alternate arrangements may be used.

The design of penetrations, described in general terms, is acceptable to us. Additional studies will be required, however, to clarify all the details of the arrangement of reinforcing bars at the openings, of the liner, and of the anchors.

3.2 Class I Structures

The applicant presented in Amendment 5 a document entitled "Ultimate Strength Criteria to Ensure No Loss of Function of Piping and Vessels under Earthquake Loading," WACP-5890, Revision 1. This document contains stress loading criteria which Westinghouse proposes as their basis for designing vessels and piping.

Our present position is that all Class I structures, systems, and components should be designed to withstand:

(a) Load combinations including normal design loads and design

earthquake loads within normal working stress or deflection limits.

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

The Class I items can be broadly subdivided into three categories: Buildings and Structures, Mechanical Systems, and Instrumentation and Control. Since Class I items are intended to perform different functions, they will require, in general, different acceptance limits under type (b) load combinations.

The seismic design criteria for Class I mechanical systems, some of which are listed below have been specifically reviewed as discussed in subsequent sections:

(a) Reactor vessel, its supports and vessel internals including fuel assemblies and control rod drives.

(b) Reactor coolant system, including piping, valves, steam generators, pressurizer, pumps and component supports.

(c) Emergency core cooling system, including piping, valves, water tanks, accumulators and pumps.

(d) Containment safeguards systems including piping, tanks, valves, ducts, fans, coolers and spray headers.

In response to our request for a definition of the proposed load combinations and stress or deformation limits, the applicant supplied information for reactor internals, vessels, piping, and supports in the Fifth Supplement, pages 19 through 52. The stress limits for type (b) loading (maximum earthquake plus pipe rupture loads) were supplied with the Fourth Supplement in the report WCAP-5890-1.

We have reviewed these submittals and we consider the loading combinations assumed by the applicant (Table 10-1) both realistic and satisfactory. The proposed stress or deformation limits for the specific components are discussed in more detail below:

3.2.1 Reactor Vessel Internals

To be able to perform their function, i.e. allow core shutdown and cooling, the reactor vessel internals must satisfy deformation limits that are more restrictive than the stress limits for other components. The applicant stated that the internals will be designed to withstand normal design loads plus earthquake loads within Section III limits, with exception of materials not covered by the Code, such as fuel rod cladding. Seismic stresses will be combined in the most conservative way and will be considered as primary stresses. We consider these criteria satisfactory.

For the type (b) loading, including maximum earthquake loads and blowdown effects due to a pipe break, the deflections are listed in Table 10-3. We consider these deflections to be reasonable. We intend to review the applicant's calculations for selected internals at the operating license stage of our review.

3.2.2 Vessels, Piping and Supports

We have reviewed the stress limits for these components, proposed by the applicant in Table 10-1 (Fifth Supplement) and the report WCAP-5890-1. We find the Section III or B31.1 Code limits, for vessels and piping respectively, satisfactory for type (a) load combination (normal design loads plus design earthquake loads).

We agree also that for type (b) load combination, (corresponding to load combination 4 in Table 10-1) the allowable extent of plastic deformation can

be larger than that associated with the Section III stress limits. We believe, however, that it would be prudent to assure that the primary stresses do not exceed the "collapse stresses" as defined in the "Criteria of Section III of the ASME Boiler and Pressure Vessel Code for Nuclear Vessels," pages 5 and 6. These primary stress limits based on plastic collapse are discussed also in ORNL-NSIC-21 "Technology of Steel Pressure Vessels for Water-Cooled Nuclear Reactors," pages 341 through 346.

The "collapse" stresses for combined primary loading have been obtained on the basis of limit design theory and perfect plasticity with no strain-hardening. The actual strain-hardening properties of specific materials, balanced to a certain extent by imperfections in the materials, will provide larger or smaller margins of safety.

Our position is also in agreement with that expressed by the "Tentative Regulatory Supplementary Criteria for ASME Code - Constructed Nuclear Pressure Vessels," which on page 29 states that where limit analysis is used the combined loadings shall be limited to 90 percent of the yield collapse load.

Since the stress limits, proposed by the applicant in WCAP-5890-1 for type (b) loading, exceed those described above, we conclude that they do not provide an adequate margin of safety. We intend to have the applicant identify specific components for which stresses under type (b) loading would exceed the "collapse" stresses used as a basis for Section III stress limits. We intend also to find out what design modifications are necessary to meet these limits.

In conclusion, it is our finding that the design method is acceptable, however the stress limits proposed for type (b) loadings should be modified

to provide an adequate margin of safety.

4.0 Core Thermal, Hydraulic, and Physics Design

The thermal-hydraulic and physics aspects of the Diablo Canyon facility were presented in our previous report to the ACRS. Since the time of that report additional information has been received on programs for fuel development, use of fixed poison rods and additional information on the use of partial length control rods. A table summarizing the important core parameters of the Indian Point II and Diablo Canyon designs is presented in Table I.

4.1 Physics Aspects

The Diablo Canyon facility physics design basis has been modified to include fixed burnable poison in the first fuel cycle. Borosilicate glass encapsulated in stainless steel rods will be distributed throughout the core in unused control rod guide tubes. It is proposed that about 1144 of these rods be installed in vacant control rod guide tubes, held in place by a spider assembly compressed beneath the upper core plate to ensure flow forces will not cause motion. These rods would have a combined worth of 7.2% delta k/k, and as a consequence the dissolved boron concentration during operation is reduced. The reduced dissolved boron concentration results in negative moderator temperature coefficients which will reduce the potential severity of loss of coolant accidents and rod ejection accidents and, according to the applicant, will damp induced xenon occillations.

The reactivity worth of the borosilicate glass rods is being evaluated at the Westinghouse Reactor Evaluation Center by comparing calculated and measured worths from critical experiments. Based on preliminary evaluation, Westinghouse has confidence in predicting the reactivity worth of the poison rods. Long term performance of these rods in a power reactor environment will

Table I

Comparison of Diablo Canyon and Indian Point II

	Diablo Canyon	Indian Point II
Total Heat Generation, Mw(t)	3250	2758
Average Heat Flux, BTU/hr ft^2	207,000	175,600
Peak Heat Flux, BTU/hr ft^2	583,000	570,800
Average Linear Heat Generation, kw/ft	6.7	5.7
Peak Linear Heat Generation, kw/ft	18.9	18.5
Core Mass velocity lb/hr ft^2	2.56×10^6	2.56×10^6
Core Inlet Temperature, °F.	539	543
Peaking Factors		
Fq	2.82	3.25
$F_{\Delta H}$	1.70	1.88
DND ratio (W-3)	1.81	1.81
Boron Concentration for Keff $\doteq$ .99 all rods out, cold, ppm	1600	3400
Moderator Temperature coefficient, $\Delta k/k$-°F	$-.5$ to -3.0×10^{-4}	$+1.0$ to -3.0×10^{-4}
Fuel Enrichments		
Region 1	2.2	2.23
Region 2	2.7	2.38
Region 3	3.3	2.68

be evaluated from in-pile testing of two rods in the Saxton reactor.

The applicant states in Amendment 7 (page II-1) that inclusion of burnable poisons will damp xenon oscillations in the X-Y plane since the moderator coefficient is negative by a sufficient margin. The threshold for X-Y instability due to feedback from the moderator temperature coefficient is calculated to be $^{-}.07 \times 10^{-4}$ delta k/k-°F. The applicant analyzed uncertainties in the variables used in the prediction of stability and has related these variables to the magnitude of moderator coefficient. The applicant believes the design moderator temperature coefficient is sufficiently negative to ensure stability.

Insofar as axial stability is concerned the applicant will install partial length rods to be moved as a bank to damp induced oscillations. The partial rods will also be used to provide flattening in the axial direction and hence the peaking factor for heat flux has been reduced from previous Westinghouse designs. Additional comments on this aspect are presented in the thermal-hydraulics section.

4.2 Thermal-Hydraulics

The core design for the Diablo Canyon reactor takes advantage of reduced peaking factors which are made possible by the use of partial length control rods. This change makes it possible to increase the average power of the core 18% compared to previous designs, yet maintain peak specific fuel powers in line with past designs. In effect, although the minimum DNB ratio in the core remains constant, the number of fuel rods which are operating close to the minimum DNBR is increased in Diablo. To illustrate this point, discussions with Westinghouse personnel have indicated the following comparisons between Diablo Canyon and Indian Point II:

	Number of rods with DNB less than indicated	
	Diablo Canyon	Indian Point II
100% power, normal flow, design inlet temperature		
DNBR of 1.8	0	0
1.9	150	<10
2.0	550	110
125% power, normal flow, design inlet temperature		
DNBR of 1.3	750	105
1.5	2500	1000
100% power, 90% of normal flow, design inlet temperature		
DNBR 1.3	0	0
1.5	0	0
1.7	250	15
1.9	1500	500
100% power, 80% of normal flow, design inlet temperature		
DNBR 1.3	0	0
1.5	550	50
1.7	2300	700

The design basis for analyzing transients in this core is that the minimum DNBR shall not be less than 1.3, and we have concluded that even though a greater number of fuel rods would be involved which approached DNB (e.g., more rods could have a calculated DNBR between 1.3 and 1.4), statistically there is ample margin of safety.

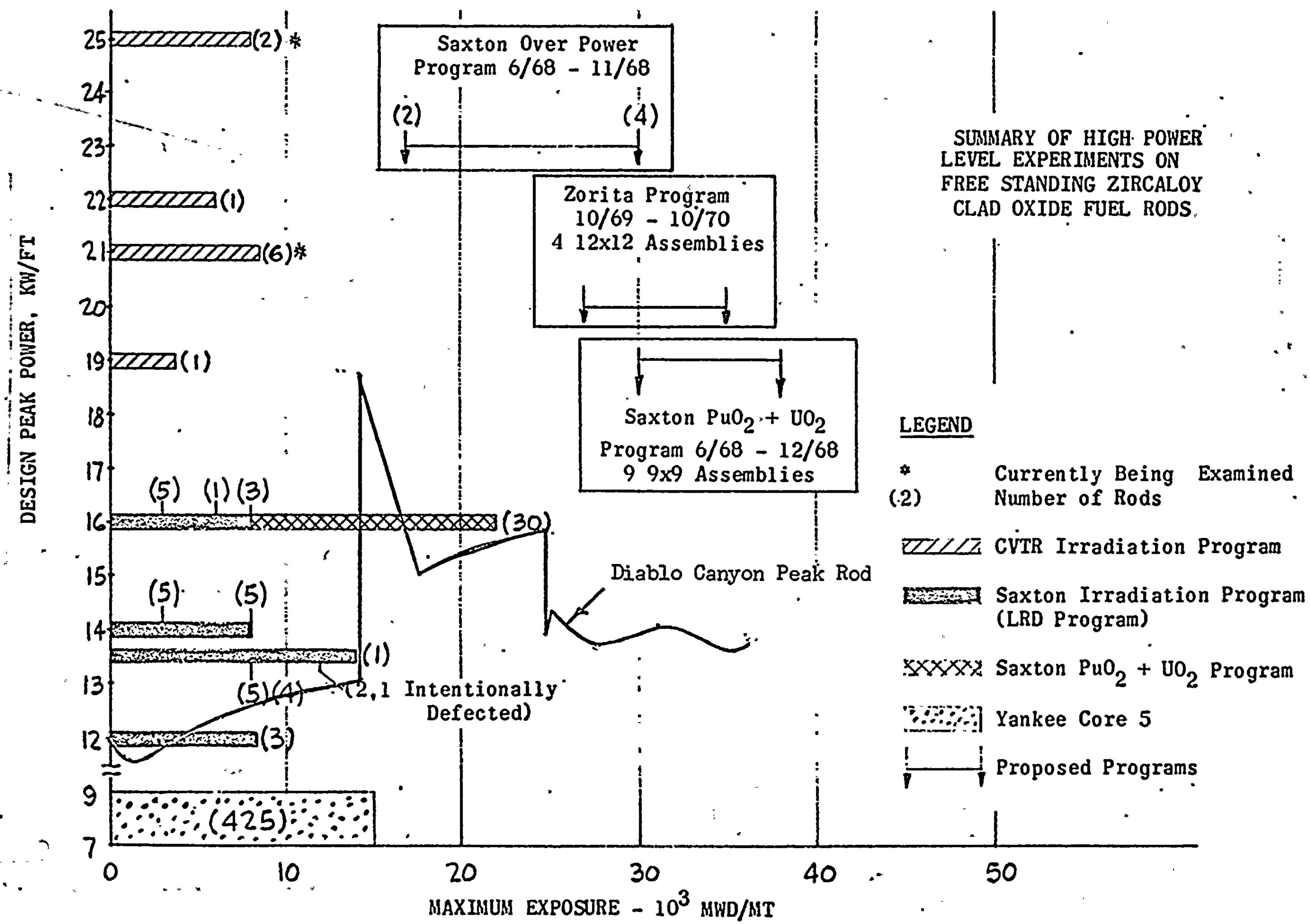
We do not agree; however, that sufficient instrumentation is being proposed to ensure that the axial flattening (peaking factors) will in practice be achieved. The applicant has proposed that reliance be placed entirely on the

four external flux monitors to detect and correct abnormal power patterns. The in-core monitors for Diablo Canyon, as presently proposed, are six traveling flux probes which may be positioned in any of 58 thimble locations in the core. These in-core channels are not designed to operate at full power for more than a few months. The applicant's position on in-core monitors is that test programs (primarily at SENA) will adequately demonstrate the capability of the external long ion chambers to predict power patterns within the core. Our position in this regard is that intelligence from in-core monitors must be provided to an operator to position the partial rods in order to assure proper axial power flattening. If, at some later date, experience shows that the external monitors will detect in-core anomalies with adequate sensitivity we would change our position.

One other aspect of our review for Diablo Canyon is that of fuel performance at proposed peak powers corresponding to expected burnup. The applicant provided a summary bar chart showing both the present and proposed irradiation test programs to demonstrate acceptable fuel performance for this reactor. We have plotted the expected peak rod operating characteristics on this bar chart. As is evident, at the present time there is no satisfactory operating experience at the linear power generation levels contemplated for the Diablo Canyon reactor. We believe, however, the test programs for Saxton and Zorita will provide a basis for predicting operation of the Diablo Canyon facility.

5.0 Instrumentation and Control

(This section under preparation and will be completed and transmitted to committee as soon as possible.)



SUMMARY OF HIGH POWER LEVEL EXPERIMENTS ON FREE STANDING ZIRCALOY CLAD OXIDE FUEL RODS

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6.0 Accident Evaluation

Accidents for the Diablo Canyon facility have been evaluated in conformance with the guidelines of Part 100.

Although our assumptions differ somewhat from those used by the applicant, all of the resulting doses, with the exception of the TID 14844 type 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 (7.5 miles) without any thyroid dose reduction factors needed.

For the loss of coolant accident which results in the TID 14844 fission product release fractions (100% noble gas, 25% iodine, and 1% solids) available for leakage, with no iodine reduction, we have calculated the following dose levels:

	2 Hour Dose (Rem) @ 0.5 mile	30 Day Dose (Rem) @ 7.5 miles
Thyroid	870	154
Whole Boyd	6	1.0

The following additional assumptions were made in calculating these doses:

1. Meteorology - Ground release, centerline, Pasquill Type F, 1 m/sec., and wake of the building (volumetric source and c = 1/2) for the first 8 hours of the accident; from 8 to 24 hours ground release, Pasquill Type F, 1 m/sec., uniform dispersion into a 22-1/2° sector; and 1 day to 30 days - the stability, wind speed, and direction were varied.

2. Leak Rate - 0.1%/day for 24 hours, and 0.045%/day for the duration of the accident.

It is apparent from the above table that an iodine reduction factor of about 3 is needed to meet the 2-hour thyroid dose limit of 300 rem at the site boundary. No reduction factors are needed to meet 30 day dose limits at the available low population zone distance.

Although the design basis for sizing the emergency core cooling system is to limit fission product release from the fuel, it has been our position that the containment and its associated engineered safety features be capable of limiting potential doses in conformance to Part 100 criteria. The applicant initially proposed a containment spray system using sodium thiosulfate to provide the needed iodine removal. In Amendment No. 2 (pages 128-130) a test program for this system was described. Results of these tests and a research and development program were further defined in Amendment No. 6 (pg. 6-7). We have discussed the proposed research and development program with the applicant, and PG&E has stated that space is being reserved near the air recirculation units so that charcoal filter units can be added in the event the research and development does not provide conclusive evidence to support needed iodine removal rates.

In addition to making independent dose calculations in conformance with Part 100 guidelines, we have reviewed the design bases for the emergency core cooling system and the containment heat removal systems. Our evaluation of these systems follows:

6.1 Design of Emergency Core Cooling Systems

The criteria for these systems as given in Amendment 2 is "that the maximum calculated zircaloy clad temperature will not at any point in the core exceed the melting temperature of zircaloy. The core will remain in its nominal heat transfer geometry and zircaloy-water reactions will be limited to an insignificant amount. The emergency core cooling system (accumulator tanks and Safety Injection System) will be designed to provide sufficient injection of borated water to meet this criterion for all reactor coolant pipe break sizes and locations up to and including a double-ended rupture of the reactor coolant pipe."

We have held meetings with the applicant with regard to the degree of redundancy required to meet the design objective given above. Our stated position to the applicant is that redundant systems should be provided such that an active component failure for both short and long term conditions and passive failure for long term cooling requirements can be tolerated without jeopardizing the ability of providing core cooling. In effect what this means is that common headers as originally proposed for safety injection and long term recirculation were not acceptable. This criterion, in our view, also applies to the component cooling water system and the auxiliary salt water system since single failures in these systems could also negate long term core cooling. In response to our interpretation of Criterion 44, the applicant modified the salt water system in Appendix A of Amendment No. 2, modified the auxiliary coolant water system in Appendix B of

Amendment No. 3, and modified the Safety Injection System in Amendment No. 7. We have reviewed these revised drawings and find certain exceptions to the desired design goal. The applicant still retains single valves which join otherwise redundant independent systems. While it may be desirable or even necessary to have the capability of transferring flow around specific components in one cooling loop and utilize components in the opposite loop, use of single valves to accomplish this objective can place both systems in jeopardy because of a single failure. If, for example, this single valve should begin to leak excessively, both recirculation systems would have to be secured to isolate the failure. Installation of dual valves in these locations would eliminate this objection. One other location we have identified where a single failure cannot be tolerated is in either of the isolation valves on the containment sump lines. We have stated our objections to the applicant and have indicated that modification during design will need to be made.

We have reviewed the performance of the Safety Injection System in being capable of meeting the design objectives. Specific answers to questions by the staff with regard to ECCS capability were made in Amendment No. 3 (pp 114-248). We have reviewed the information submitted and believe the system as proposed is generally adequate (the specific area of thermal shock is still under review). The performance of the system with 3 of 4 accumulators and 1 of 2 S.I.S. pumps can be summarized as follows:

	Break Size	Maximum Cladding Temperature, °F	Percent Metal-Water Reaction
1.	Double-ended coolant pipe break	2120	<1
2.	3.0 ft^2	1615	0
3.	0.5 ft^2	1795	0

Study of the problem of thermal shock during core cooling system actuation by Babcock & Wilcox, Combustion Engineering, General Electric and Westinghouse continues. Two modes of potential failure are being considered: ductile yielding and brittle fracture. The latter is being treated using both the Pellini-Puzak diagram approach and fracture mechanics.

We are presently waiting for the results of calculations, promised by Westinghouse in a topical report, to establish thermal stress distribution patterns near the crack tip as the crack progresses through the thickness of the vessel. Since the information submitted by Westinghouse, so far, in connection with the Diablo Canyon application is insufficient, we consider the thermal shock problem unresolved at this time.

6.2 Engineered Safety Features for Heat Removal from the Containment

The Diablo Canyon Containment vessel is designed for an accident pressure of 47 psig. The applicant was asked to perform calculations to show the capability of the containment to withstand various assumed energy releases during the course of an accident. The answers to these questions appear in Amendment No. 3 (pp 181-219). Engineered Safety Features for the containment structure are redundant and the applicant's analysis shows that

operation of 3 of 5 containment air coolers and 1 of 2 containment spray systems is adequate to maintain the calculated pressure below design pressure. We have reviewed the accident model and have concluded that the containment and its heat removal systems are adequately sized.

One aspect which we believe needs further attention during detailed design is that of leak detection on external recirculation systems. The recirculation features are closely associated with the ECCS (for long term heat removal) and our concern is that of detecting and being capable of isolating leaks in either of the two systems. If the leakage from valves and packings are within design limits, the dose contribution can be tolerated within Part 100 guidelines. If major leaks should develop during the recirculation phase, activity leakage to the environment (there is no provision for iodine removal in the auxiliary building ventilation system) could become excessive unless an operator has provision to detect and isolate the source.

6.3 Control Room Shielding

The accident dose criteria for this control room (including ingress and egress) is 2.5 rem whole body and 300 rem to the thyroid for the course of an accident. In our opinion, an iodine removal system should be incorporated into the control room ventilation system or other measures should be taken to limit potential thyroid doses in the control room to values more in line with Criterion 11 considerations.

7. Conclusions

Assuming satisfactory resolution, as the final design evolves, of specific problems enumerated in the foregoing sections, we have concluded that there is reasonable assurance the Diablo Canyon facility can be built and operated at the proposed location without undue risk to the health and safety of the public.

APPENDIX A

LIST OF AMENDMENTS - DIABLO CANYON FACILITY

1. Amendment No. 1 dated July 10, 1967 which contained answers to questions, design methods based on ultimate strength criteria, and described part length absorber rods.
2. Amendment No. 2 dated July 24, 1967 which contained answers to questions.
3. Amendment No. 3 dated July 31, 1967 which contained answers to questions and additional information on site geology.
4. Amendment No. 4 dated September 8, 1967 which provided a cross reference to pages in the PSAR which dealt with each of the proposed General Design Criteria.
5. Amendment No. 5 dated October 18, 1967 which contained financial data, additional tsunami information and revised information on the ultimate strength design criteria.
6. Amendment No. 6 dated November 6, 1967 which contained answers to questions, outlined research and development programs, and presented topical reports on the use of burnable poison rods and experimental results on DNB studies in rod bundles.
7. Amendment No. 7 dated November 9, 1967 which contained answers to questions.