SUPPLEMENTAL
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OCT 26 1967

Docket No. 50-275

Pacific Gas & Electric Company
245 Market Street
San Francisco, California 94106

Attention: Mr. Richard H. Peterson
Senior Vice President &
General Counsel

Gentlemen:

This refers to your application for a construction permit and facility license for a nuclear power plant to be located at the Diablo Canyon site. We have reviewed Supplements 1, 2 and 3 to your application which were submitted in response to our questions for additional information. Further discussions on the proposed plant design were held with representatives of your company, the Westinghouse Electric Company, the Division of Reactor Licensing, and the Advisory Committee on Reactor Safeguards on October 3, 4, and 5, 1967.

As we have discussed with you, we consider the following major safety review areas as not completely resolved in terms of satisfactory documentation of the plant design and proposed operation:

(a) seismic design in terms of criteria for all Class I systems and components considering load combinations and allowable stresses or deformation.

(b) the instrumentation and control system in regard to the proposed increased core average power density including an evaluation for additional in-core monitoring capability in conjunction with the part-length control rods.

(c) a safety evaluation to support the adequacy of the proposed emergency core cooling system including consideration of the existence of single pipe lines connecting various subsystems.

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(d) a safety evaluation associated with the effects of seismic and blowdown forces acting on the reactor core to affect reactivity.

We understand that you intend to submit additional information in November 1967 in response to our letter, dated August 31, 1967. This information relates to the research and development programs appropriate for the Diablo Canyon plant and the appropriate matters identified by an asterisk in the June 15, 1967 ACRS report on the Vermont Yankee Nuclear Power Station.

We also believe that supplemental information related to a preliminary design and safety evaluation of the instrumentation and control system is required. The description and evaluation of the protection systems, submitted in the second and third supplements, which actuate reactor trip, containment isolation, emergency core cooling system, and other engineering safety features do not provide a sufficient technical basis upon which to make a safety evaluation. An evaluation of the protection systems based on this information would be little more than an evaluation of design criteria. We need the specific information requested in our questions, including a preliminary design, so we can evaluate the approach to implementation of the criteria.

In addition to this information, based on discussions at the recent meetings and review of the information available, we have found that further information is required to enable us to continue and complete our review. In this regard, we require the specific information detailed in the enclosed questions.

To complete our review and to prepare a report to the ACRS, we will require early submittal of complete and satisfactory replies to this request. We shall be available to discuss and clarify any of the aspects of the foregoing with you.

Sincerely yours,

/s/

Peter A. Morris, Director
Division of Reactor Licensing

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AEC Doc. Rm.
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J. R. Buchanan, ORNL
S. Levine
H. Steele

Enclosure:
Request for Additional Information

SEE ATTACHED SHEET FOR OTHER CONCURRENCES

cc: Mr. J. R. Stadelman
Westinghouse Atomic Power Division
P. O. Box 355
Pittsburgh, Pennsylvania 15230

AIRMAIL

OFFICE ▶	DRL:RP	DRL:RT	DRL
SURNAME ▶	RLTedesco/dj RSBoyd	SLevine	PAMorris
DATE ▶	10/26/67	10/ /67	10/26/67

REQUEST FOR ADDITIONAL INFORMATION

PACIFIC GAS & ELECTRIC COMPANY

DOCKET NO. 50-275

1. We understand that modifications are to be made to the reactor protection system to use a differential signal derived from the upper and lower half sections of the long ion chambers to reset the level trip setting. In this regard, please describe the proposed changes including a safety evaluation and a preliminary design covering the significant aspects of the modifications.

2. Describe the results of a safety evaluation treating the effects of forces on the reactor core considering the movement of fuel rods and fuel pellets from damaged fuel rods. The evaluation should be broad in scope to include the superposition of seismic loads and loss of coolant accident loads as well as each of these loading conditions acting independently. Areas of interest include the reactivity effects due to fuel movements and the ability of the core to remain intact and coolable under these conditions.

3. Our preliminary review of the piping systems proposed for the emergency core cooling system indicates the existence of single pipe lines connecting the various subsystems from the refueling water storage tank to the Auxiliary Building and then through the containment to the reactor primary system with a return via the recirculation system. Such an arrangement suggests a potential loss of core cooling capability in an unlikely event of any failure leading to excessive leakage from the proposed single line system. On this basis, please describe the protection provided in terms of leakage detection and isolation capability that will be provided to detect such failures to assure core cooling.

4. Our Design Criteria, which were published for comment on July 11, 1967, includes a criterion (number 44) for providing at least two independent core cooling systems. Our review of the proposed emergency core cooling indicates the existence of single subsystems (even though redundancy has been provided in certain of the active components as well as for the accumulators) to provide core cooling capability over the entire primary system piping break spectrum. On this basis, please describe in detail the results of your safety evaluation of the proposed core cooling system in consideration of meeting the objectives stated in Criterion 44.

5. Provide a detailed sketch showing the piping configuration from the containment sump to the RHR system as indicated in Figure VI D 3 of Supplement 3. Emphasis should be given to the guard pipe design leading from the containment sump to Auxiliary Building.

6. The proposed diesel fuel storage capacity as stated on page 135 of Supplement 3 is 5 days. Describe the evaluation conducted that leads to the conclusion that a 5-day supply is adequate. Include a discussion related to obtaining additional fuel from nearby sources.

7. Describe the results of your safety evaluation of the containment penetration design for the main steam lines. Indicate the design criteria for these penetrations which should include the local effects of jet impingement and thrust loads for assurance of maintaining containment integrity.

8. Describe the results of a safety evaluation regarding the safety aspects of a seismic scram trip for the proposed facility.

9. Describe the design provisions including a safety evaluation regarding the capability of control rod cluster insertion to make the reactor subcritical considering the combined loading effects resulting from earthquake and blowdown forces. Include a discussion concerning provisions in the design of the reactor internals to assure that the control rods would be inserted into the core by gravitational forces under combined effects of seismic and accident forces.

10. For the piping, vessels, supports, and reactor vessel internals, separately, provide the following criteria type information:

    (a) the loading combinations to be employed in the designs and the reasoning for selection of the loading combinations

    (b) the stress or deformation limits for each case (a)

    (c) the margin of safety between (b) and the expected collapse or failure condition.

11. For particular cases where calculations of maximum limits of deformation or stress (at which inability to function occurs) have been made, please supply such data. Also in such cases supply the maximum design limit value, and the expected deformation or stress where this has been calculated. In all such cases the applicable loading combination should be identified, and any amplifying discussion of the margin of safety presented.

12. In cases where only stress intensity criteria are to be employed, as outlined in WCAP-5890-1, provide a realistic estimate of the strain, based on appropriate material properties at the applicable temperature, in order that an estimate of the margin of safety can be made. Especially for those cases which fall outside the code-based "boxes" and in the region between the boxes and the limit curves it is imperative that the margin of safety

be identified. Confirmatory test data should also be supplied, if available, with interpretative discussion. With regard to WCAP-5890-1, provide additional discussion regarding the applicability of the derivations in the report to the strain hardening domain.

In connection with the above, provide information that will permit evaluation of the effect of welds, irradiation, corrosion, material imperfections (flaws), etc., in the design approach.

13. To the extent available, supply criteria or specific information on the interaction forces, deformation and stresses connected with the relative motions between the reactor vessel, steam generators or other large components. Indicate how these relative motions will be controlled by snubbers or other means, and what reaction forces (and corresponding stresses) will be transmitted to the pipes.

14. The containment structure will be founded on bedrock and surrounded by some 30 feet of overburden. Describe how the forces arising from any soil-structure interaction arising from normal and seismic loading will be evaluated and handled in the design.

15. Provide the actual and the simplified stress-strain curves for:

    (a) The two materials used in recent limit analysis tests performed by Westinghouse.

    (b) The piping, vessel, and internals materials for which you intend to use the limit analysis method outlined in WCAP-5890-1.

    For materials listed in (b) above, provide also, if available, the stress-strain curves obtained from tests using larger test specimens than those specified by the ASME. Discuss the thickness effect on the shape of the stress-strain curve, and estimate the maximum material thickness for which the limit curves given in the WCAP-5890-1 are applicable.