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IN REPLY REFER TO:
Docket No. 50-275

MAY 1 8 1967

Pacific Gas and 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 dated January 16, 1967, for a construction permit and facility license which would authorize construction and operation of a nuclear power reactor at the Diablo Canyon site located in San Luis Obispo County, California.

On March 21, 1967, and April 20-21, 1967, members of the regulatory staff met with representatives of your company to discuss various aspects of the plant design. As a result of this meeting, we requested additional information pertaining to the site, plant layout, and containment structural design by letter dated May 5, 1967. We indicated in the referenced letter that questions related to other aspects of the design would be forwarded in subsequent correspondence.

Accordingly, you are requested to provide the information listed in the enclosure pertaining to instrumentation, control, and power systems. We are continuing our review and will develop further questions in the remaining areas of the plant design.

Your reply to these attached questions should be submitted as an amendment to your application. The staff, of course, will be available as may be required to discuss and amplify the meaning of the questions.

Sincerely yours,

ORIGINAL SIGNED BY
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
Request for Additional Information

AIRMAIL

DRL/RP	DRL/RP	DRL/RT	DRL/RP	DRL
KWoodard/dj	RLTedesco	SLevine	RSBoyd	PAMorris
5/17/67	5/18/67	5/18/67	5/18/67	5/18/67

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REQUEST FOR ADDITIONAL INFORMATION

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON REACTOR

DOCKET NO. 50-275

IV. Instrumentation, Control, and Power Systems

A. More specific design criteria should be provided for the protection systems which initiate reactor trip, containment isolation, emergency core cooling, and other engineered safety features. For example, each of the items of section four of the IEEE proposed Standards for Reactor Protection Systems (Rev. 7 or 8) should be addressed. For those items of the proposed standard which are criteria for Diablo Canyon, discuss how the criteria are to be met. If any are not to be met, discuss why they are not necessary for this design.

B. Please compare and identify any differences between the reactor protection system and the instrumentation and controls for engineered safety features for Diablo Canyon and those for the H. B. Robinson and Point Beach reactor plants.

C. Please supply elementary (schematic) diagrams of the logic circuits which actuate reactor trips, emergency core cooling, containment isolation, and other engineered safety features.

D. Identify and describe the control functions (in addition to actuation) required for successful operation of the engineered safety features.

E. Describe each of the reactor trip channels more fully, including:

1. Type of components (sensors, amplifiers, bistables, etc.).

2. Trip point arrangement fixed, changed by mode switch, changed by another variable, and means for bypassing.

3. Where variable trip points are used, indicate whether the means for varying the trip point meet the single failure criterion.

4. With regard to testing, state the design criteria and indicate:

(a) what parts of the circuit can be tested at power

(b) what parts of the circuit must be tested when shut down

(c) how does the test check for the loss of redundancy

(d) where and how is the test signal injected

(e) how is the result of the test detected.

F. Describe specifically the interlocking of reactor trip signals with measured nuclear and steam power (page 7-3).

G. Analyze the 'rod-drop' protection in terms of the assurance there is that 'out-of-core' neutron detectors will detect the dropping of any rod (consider the case for 1 of the 4 neutron channels out of service). Describe the protection circuits specifically including features to meet the single failure criterion and how they will be tested.

H. Does the 'low-rod-insertion' limit alarm (page 7-4) meet the single failure criterion? Describe the circuit.

I. Describe the local stations where the operation of engineered safety features and plant shutdown equipment can be controlled and monitored in case the central control room becomes uninhabitable. Where are the stations located and how is the operator protected from excessive radiation? What indications of plant status are provided? Include a discussion of the design features to be considered for transfer of operations from the central control room to the local stations.

J. What instrumentation is required for post accident recovery operations? State your criteria for indicating to the operator the reactivity status of the reactor and the pressure, temperature, water levels, hydrogen, and boron concentration in the containment following an accident.

K. Evaluate the ability of the reactor protection equipment and electrical equipment for the engineered safety features to withstand the environment in which it must operate. The evaluation should include but not be limited to the following:

1. Identification of the equipment, including cables, inside the containment which must function in an accident environment. State the environment and expected length of time for which this equipment must function. Describe the tests or test data which will demonstrate the ability of the component to function in the accident environment.

2. An evaluation of the performance of the control room equipment under abnormal conditions, such as heating or air conditioning failure.

3. Is forced cooling required for any of the sensors? What are the effects of loss of forced cooling? Is the cooling monitored?

4. An evaluation of the ability of the electrical equipment such as pump motors, cables, and instrumentation in the emergency core cooling systems to withstand the radiation to which it will be exposed during recirculation after a major accident.

5. A description of the containment sump level instrumentation and a discussion of their capabilities in terms of environmental conditions and maximum water levels.

L. Are the ECCS accumulator level sensors connected to common sensing lines?

M. What prevents improper operation of the safety injection block switch?

N. Discuss the use of the coincidence of pressurizer level and pressure for the actuation of emergency core cooling. Is there no condition in which emergency core cooling might be required when either low pressure or low level might not be sensed? Could the reference leg of the level sensors be voided as the result of the blowdown? Are spurious trips from the pressure or level instruments a serious enough threat to justify a system which requires tripping both for ECCS actuation?

O. Do the containment high activity and high pressure signals which initiate purge valve closure each meet the single failure criterion? Is each channel of these systems continuously recorded? Describe the type and location of local indications.

P. Evaluate the ability of the rod position indication to provide the operator with continuous information on the reactivity status of the core. For example, what information is indicated in the control room by means of readouts, indicator lights, print outs, etc.? To what extent do the pulse counter digital readouts back up the readouts from the differential transformer transmitters? What errors might an operator be led to make as the result of a single failure in the rod position indication?

Q. The maximum number of RCC assemblies which can be moved and their speed will be determined by detail plant design. If the nuclear power level trip is unable to protect against the simultaneous withdrawal of all RCC assemblies, will interlocks be provided to prevent simultaneous withdrawal? Will the interlocks meet the single failure criterion?

R. Evaluate the ability to supply electric power to engineered safety features under accident conditions from the incoming power lines. The evaluation should include but not be limited to the effect of abrupt loss of the Diablo Canyon plant, fault on the incoming lines, and faults and equipment failures in the plant or substation. Specifically address the problems associated with the use of a single startup transformer. Are the buses containing the engineered safety features (buses number 2 and 3) normally supplied from the startup transformer or must they be transferred from the main unit to the startup transformer under accident conditions? Describe the motor operated breaker in the main bus and describe the signal for operation and time-to-open characteristics.

S. Evaluate the ability of the electric system to supply power for emergency core cooling and other engineered safety features under conditions of a major reactor accident with concurrent loss of off-site power and a single failure in the on-site electric system. The evaluation should include but not be limited to the following:

1. A specific description of the sequence after a major accident with a loss of external power. State which circuit breakers are open and which closed, the sequence of operations, the time intervals involved, which buses are isolated, which buses are connected together, which generators supply each bus, and the instrumentation and relaying which initiates the various operations.

2. Same as (1) except with a single failure in the on-site electric system. Include, but do not limit the discussion to failure of any diesel engine to start, failure of a supply or load circuit breaker, sequencer failure, bus fault, and failure in the d-c control circuit. Specifically, address the question of whether reliability is lost by connecting the engineered safety features to buses which also supply other loads.

3. Ratings of the diesel generators and the station batteries.

4. List each component of the engineered safety features and its load requirement.

T. Evaluate the ability to provide power to engineered safety features from off-site and on-site sources with any single failure in the d-c system which supplies control power for circuit breakers, valves, etc. Discuss more specifically how the d-c buses are arranged to supply alternate power sources for systems where redundancy is employed.

U. Describe the switching sequence to energize the boric acid pumps and the motor driven emergency feedwater pumps in the event of the loss of off-site power, including design provisions to meet the single failure criterion.