D. J. Skovholt, Assistant Director for Reactor Operations

January 15, 1968

Thru : Saul Levine, Assistant Director for Reactor Technology

V. A. Moore, Chief
Instrumentation & Power Technology Branch, DRL

SAFETY EVALUATION OF INSTRUMENTATION AND POWER, HUMBOLDT BAY POWER PLANT, DOCKET NO. 50-133

DRL:I&PTB:DFS RT-315

By letter dated December 15, 1967, the Pacific Gas and Electric Company responded to several staff questions relating to Instrumentation and Power.

The safety evaluation of this information has been performed by D. F. Sullivan, and is herewith attached for inclusion in the report to ACRS for consideration at the February meeting.

Attachment:
Safety Evaluation

cc: D. L. Ziemann, DRL
D. F. Sullivan

bcc: V. A. Moore, DRL

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SURNAME ▶	VAM V. A. Moore D. F. Sullivan	SL for S. Levine	
DATE ▶		1-15-68	

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Form AEC-318 (Rev. 9-53)

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PACIFIC GAS AND ELECTRIC COMPANY,
HUMBOLDT BAY POWER PLANT, DOCKET NO. 50-133

Safety Evaluation

1) The applicant was essentially unresponsive to our request for a detailed failure mode analysis of the reactor protection system using the IEEE Standard (Rev. 8 or 9) as a guide. His position is that the expression "single failure" has not been defined in sufficiently precise terms to permit such an analysis.

While granting that the various standards groups are pursuing a continuing study of this matter for the purpose of formalizing the pertinent definitions, it is also true, in our judgment, that a sufficient understanding of "single failure" does exist throughout the industry which can allow an analysis, sufficient for our purposes, to be conducted. Such analyses are commonplace in SAR's. Further, a completely satisfactory analysis of this kind has been performed by the BONUS reactor staff.

Accordingly, we must conclude that the applicant's response is unacceptable.

We believe that the applicant should perform the requested analysis and accomplish the system modifications required to meet the single failure criterion.

2) The applicant was requested to discuss the reliability of those power generation sources and the associated circuitry which will provide emergency power in the event of an accident and simultaneous loss of the external grid. The applicant has complied with our request and has submitted the requested discussion (essentially an analysis).

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We have reviewed the information submitted, and have concluded that neither the off-site nor on-site emergency power systems conform to criterion No. 39 even though there are multiple power sources available. With reference to their Figure 1-1, it can be seen that all off-site power to the Reactor Feed Pumps (High Pressure Injection) and the Core Spray Pumps (Low Pressure Injection) must be transmitted via Unit Three's 2400 kv bus. This bus can be faulted by several single failures. A dead short at the bus itself is an obvious example. Failure of the circuit breaker just downstream of House Transformer No. 3 to open, or failure of ACB 152-305 to close will negate the bus. In addition, both of these breakers, in addition to those which connect the individual safeguard loads to the bus, depend upon a single d.c. source (Unit No. Three's battery) for their operation. Finally, a single undervoltage relay senses voltage loss at the 2400 kv bus.

The on-site power system consists of a single 60 kw propane-driven generator.

Reactor Technology will, later this week, report orally to Reactor Operations regarding the acceptability of the total emergency power system.

3) The applicant was asked to describe the refueling interlock system and the protection (against the "refueling accident") which it provides.

If the master reactor switch is in the "refuel" position, an interlock prevents the withdrawal of more than one control rod. With the switch in

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either the "refuel" or "critical test" position, the following interlocks are in effect:

a) The reactor shuffling winch is prevented from operating in the lowering direction unless all control rods are fully inserted. (Note: this does not prevent a fuel bundle from <u>dropping</u> into the core when more than rod is withdrawn.)

b) The reactor safety system is de-energized (scrammed), and rod withdrawal is prevented if the refueling building crane is moved into the area over the reactor.

From our review of this and the remainder of the submitted information, we have concluded the following:

1) An interlock, designed to protection system standards,* should be installed to prevent the shuffling winch from positioning fuel above the core when more than one rod is withdrawn.

2) The applicant should determine if the circuits which scram the reactor when the refueling building crane is moved above the core have been designed in accordance with protection system criteria. Suitable modifications should be made if the circuits are found to be deficient.

*Current practice allows a non-redundant interlock when the mode switch is placed in the "critical test" position.

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