SEP 18 1984

Docket Nos.: 50-275
and 50-323

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Docket File 50-274/323
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Dear Mr. Schuyler:

Subject: Request for Additional Information NUREG-0737 - II.D.1. Performance Testing of Relief and Safety Valves

We have evaluated the information provided in your submittal of January 23, 1984 on the subject, in response to our request of November 15, 1983. Based on our evaluation we find that some additional information, as listed in the enclosure, is necessary for us to complete our review. We request that you provide the information within 60 days of receipt of this letter.

Sincerely,

ORIGINAL SIGNED BY

George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

Enclosure:
As stated

cc: See next page

DL:LB#3
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9/14/84

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9/17/84

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Diablo Canyon Unit 1 and Unit 2
Additional Requests for Information on NUREG-0737, Item II.D.1
Performance Testing of Relief and Safety Valves

1. The Diablo Canyon FSAR (Chapter 15.4.2.2, Amendment 5 dated March 1974) indicates greater liquid water relief through the safety valves (SVs) for the feedwater line break (FWLB) accident than that indicated in the EPRI Westinghouse inlet condition report (EPRI NP-2296). The FSAR also indicates water relief for a significantly greater time period than that of any of the EPRI tests. Identify the applicable liquid flow conditions that the SVs would be exposed to, i.e., pressure, temperature, flow rate, number of actuation cycles, and total length of time for liquid flow, for the FWLB accident and provide EPRI or other test data that demonstrate that the Diablo Canyon SVs can perform their pressure relief function and the affected unit can be safely shut down.

2. NUREG-0737, Item II.D.1 requires that the plant specific PORV control circuitry be qualified for design basis transients and accidents. Please provide information which demonstrates that this requirements has been fulfilled.

3. In one of the EPRI loop seal steam tests on the Crosby 6M6 safety valve (Test 1419), the valve chattered after opening and the test was terminated after the valve was manually opened to stop the chattering. This test had a 350°F loop seal, which is representative of the hot loop seal at both units of the Diablo Canyon Plant. Additionally, the ring settings used in this test were evidently representative of the settings used in the plant valves. Demonstrate that the behavior exhibited in this test is not indicative of the expected behavior of the Diablo Canyon valves.

4. Your response of January 23, 1984 to the NRC's request for further information does not provide values for the expected plant backpressure. Identify the expected plant backpressures for the safety valves and PORVs for steam and liquid conditions so that backpressures measured in the EPRI tests can be compared with the Diablo Canyon plant backpressures.

5. Your response of January 23, 1984 to Question 4 of our request for further information states that the Westinghouse Owners Group performed an analysis to determine the effects of blowdowns exceeding 10% on safety valve performance. The response states that results of the analysis showed no adverse effects on plant safety and that a discussion of the blowdown analysis is contained in the Westinghouse report WCAP-10105. In reviewing this document, however, we are unable to confirm that excessive blowdowns are discussed therein. Therefore, provide documentation that discusses the analysis on the effects of increased safety valve blowdowns.

6. Your response of January 27, 1984 to Question 16 of our request for further information states that supports with a natural frequency of 20 Hz or greater are modeled as rigid in accordance with Diablo Canyon licensing commitments. The use of 20 Hz in the licensing commitments as a measure of rigidity was based, however, on seismic loads. The fluid transient loads have higher frequency content and could excite higher frequencies in the supports. Thus, provide an evaluation of the effects of modeling supports having natural frequencies of 20 Hz as rigid when fluid discharge loads are imposed on the system.