Docket No. 50-344

June 12, 1985

DLR
OIC

Mr. Bart D. Withers
Vice President - Nuclear
Portland General Electric Company
121 S. W. Salmon Street
Portland, Oregon 97204

Dear Mr. Withers:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - NUREG-0737, ITEM II.D.1

The staff is continuing its review and evaluation of PGE's submittals for TMI Item II.D.1 of NUREG-0737. During the course of our review, we have identified the need for additional information concerning the selection of transients and valve inlet and downstream conditions, valve operability, and the thermal hydraulic and structural analyses of the inlet and discharge piping. It is requested that you provide your response to the enclosed Request for Additional Information within 30 days of receipt of this letter or provide a date that you can meet within 15 days of your receipt of this letter.

Should you have any further questions concerning the enclosed request, please contact the Project Manager, Lisamarie Lazo at (301) 492-7791.

This request for information affects fewer than 10 respondents; therefore, OMB clearance is not required under P. L. 96-511.

Sincerely,

/s/

Edward J. Butcher, Acting Chief
Operating Reactors Branch No. 3
Division of Licensing

Enclosure:
Request for
Additional Information

cc w/enclosure:
See next page

Distribution:

Docket File	HThompson	EJordan
PKreutzer	LLazo	BGrimes
NRC & L PDRs	OELD	PMcKee
ACRS 10	Branch Files 1+1	

ORB#3:DL	ORB#3:DL	ORB#3:DL
PKreutzer	LLazo;ef	EButcher
6/7/85	6/10/85	6/11/85

8506180597 850612
PDR ADOCK 05000344
P PDR

Mr. Bart D. Withers
Portland General Elctric Company

Trojan Nuclear Plant

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Trojan Nuclear Plant
Post Office Box 0
Rainier, Oregon 97048

Robert M. Hunt, Chairman
Board of County Commissioners
Columbia County
St. Helens, Oregon 97501

William T. Dixon
Oregon Department of Energy
Labor and Industries Building
Room 111
Salem, Oregon 97310

Regional Administrator
USNRC, Region V
Office of Executive Director for Operations
1450 Maria Lane, Suite 210
Walnut Creek, California 94596

REQUEST FOR ADDITIONAL INFORMATION

TMI ACTION NUREG-0737 (II.D.1)

FOR

TROJAN

DOCKET NO.: 50-344

QUESTIONS RELATED TO THE SELECTION OF TRANSIENTS AND VALVE INLET AND DOWNSTREAM CONDITIONS:

1. The Trojan plant specific submittal stated that the cold overpressure event was not represented by the Westinghouse 4-loop reference plant cold overpressure event. Provide additional detail and discussion on how the Trojan cold overpressure inlet fluid conditions were determined.

2. In valve operability discussions on cold overpressurization transients the submittal only identifies conditions for water discharge. According to the Westinghouse valve inlet fluid conditions report, the PORVs are expected to operate over a range of steam, steam-water, and water conditions because of the potential presence of a steam bubble in the pressurizer and water solid operations. To assure that the PORVs operate for all cold overpressure events, discuss the range of fluid conditions expected for the expected types of fluid discharge and identify the test data that demonstrates operability for these cases. Since no low pressure steam tests were performed for the PORVs, confirm that the high pressure steam case for both opening and closing of the PORVs.

3. Results from the EPRI tests on the Crosby safety valves indicate that the test blowdowns exceeded the design value of 5% for both the "as installed" and "lowered" ring settings. The Trojan submittal stated that blowdowns in excess of the ASME specified design 5% blowdown was not considered significant from a nuclear safety standpoint. If the expected blowdowns for Trojan exceed 5%, the higher blowdowns could cause a rise in pressurizer water level such that water may reach the safety valve inlet line and result in a steam-water flow situation. Also the pressure might be sufficiently decreased such that adequate cooling might not be achieved for decay heat removal. Provide additional discussion concerning these consequences and justification that nuclear safety is not impaired.

QUESTIONS RELATED TO VALVE OPERABILITY

4. The EPRI Inlet Fluid Justification Report suggested a method for demonstrating safety valve stability. This method compares the total inlet piping pressure drop for the in-plant safety valves and piping to the applicable EPRI test safety valve piping combinations. The total inlet piping pressure drop is composed of a frictional and acoustic wave component evaluated under steam conditions. The Trojan submittal did not provide pressure drop calculations but only compared piping lengths. Provide the necessary pressure drops for the Trojan expected inlet conditions, flow capacity, ring settings, and inlet piping configuration. Make a comparison with the applicable EPRI test pressure drops to demonstrate valve stability.

5. The Westinghouse inlet fluid conditions report stated that liquid flow could exist through the PORV for the FSAR feedline break event and the extended high pressure injection event. Liquid PORV flow is also predicted for the cold overpressurization event. These same flow conditions will also exist for the Block Valve. The EPRI/Marshal Block Valve Report did not test the block valves with fluid media other than steam. The Westinghouse Gate Valve Closure Testing Program did include tests with water; however, the information presented in the report did not provide specific test results. Since it is conceivable that the EMOV could be expected to operate with liquid flows, discuss EMOV block valve operability with expected liquid flow conditions and provide specific test data.

6. Bending moments are induced on the safety valves and PORVs during the time they are required to operate because of discharge loads and thermal expansion on the pressurizer tank and inlet piping. Make a comparison between the predicted plant moments with the moments applied to the tested valves to demonstrate that the operability of the valves will not be impaired.

7. The Westinghouse Valve Inlet Fluid Conditions Report states that liquid discharge could be expected through the safety valves for both the feedline break and extended high pressure injection events. The EPRI 6M6 test safety valve experienced some chatter and flutter while discharging liquid at certain ring settings. Testing was terminated after observing chattering to minimize valve damage. Inspection revealed some valve damage which was presumably caused by the valve chatter and flutter. Liquid discharge for Trojan may conceivably occur for longer periods of time than the EPRI testing. Thus, longer periods of valve chattering may cause severe valve damage. Discuss the implications this may have on operability and reliability of the Trojan safety valves. Identify any actions that will be taken to inspect for valve damage and assure reliable operability following safety valve lift events.

8. 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 requirement has been fulfilled.

9. The Trojan plant safety valves are Crosby 6M6 and were tested by EPRI. EPRI testing of the 6M6 was performed at various ring settings. The submittal did not provide the present ring settings but stated the rings were set at the factory recommended settings. If the plant current ring settings were not used in the EPRI tests, the results may not be directly applicable to the Trojan safety valves. Identify the Trojan safety valve ring settings. If the plant specific ring settings were not tested by EPRI, explain how the expected values for flow capacity, blowdown, and the resulting back pressure corresponding to the plant-specific ring settings were extrapolated or calculated from the EPRI test data. Identify these values so determined and evaluate the effects of these values on the behavior of the safety valves.

QUESTIONS RELATED TO THE THERMAL HYDRAULIC ANALYSIS OF THE INLET AND DISCHARGE PIPING:

10. The submittal states that a hydraulic analysis of the safety/relief valve piping system has been conducted. To allow for a more complete evaluation of the methods used and the results obtained from the thermal hydraulic analysis, provide additional discussion on the thermal hydraulic analysis that contain at least the following information:

    a. An explanation of the method used to treat valve resistances in the analysis. Report the valve flow rates that correspond to the resistances used. Because the ASME Code requires derating of the safety valves to 90% of actual flow capacity, the safety valve analysis should be based on flows equal to 111% of the valve flow rating, unless another flow rate can be justified. Provide information explaining how derating of the safety valves was handled and describe methods used to establish flow rates for the safety valves and PORVs in the analysis.

    b. A discussion of the sequence of opening of the safety valves that was used to produce the worst case loading conditions.

    c. Copies of the EDS Nuclear, Inc. thermal analysis reports.

QUESTIONS RELATED TO THE STRUCTURAL ANALYSIS OF THE INLET AND DISCHARGE PIPING:

11. The submittal states that a structural analysis of the safety and relief valve piping system has been conducted. To allow for a more complete evaluation of the methods used and results obtained from the structural analysis, please provide reports containing at least the following information:

    a. The REFORC 2 program was identified as the program used in the analysis. How was the program verified?

    b. A description of methods used to model supports, the pressurizer and relief tank connections, and the safety valve bonnet assemblies and PORV actuator.

    c. An identification of the load combinations performed in the analysis together with the allowable stress limits. Differentiate between load combinations used in the piping upstream and downstream of the valve. Explain the mathematical methods used to perform the load combinations, and identify the governing codes and standards used to determine piping and support adequacy.

    d. A sketch of the structural model showing lumped mass locations and pipe sizes.

    e. Copies of the EDS Nuclear, Inc. structural analysis report.

12. According to results of EPRI tests, high frequency pressure oscillations of 170-260 Hz typically occur in the piping upstream of the safety valve while loop seal water passes through the valve. An evaluation of this phenomenon is documented in the Westinghouse report WCAP 10105 and states that the acoustic pressures occurring prior to and during safety valve discharge are below the maximum permissible

pressure. The study discussed in the Westinghouse report determined the maximum permissible pressure for the inlet piping and established the maximum allowable bending moments for Level C Service Condition in the inlet piping based on the maximum transient pressure measured or calculated. While the internal pressures are lower than the maximum permissible pressure, the pressure oscillations could potentially exite high frequency vibration modes in the piping, creating bending moments in the inlet piping that should be combined with moments from other appropriate mechanical loads. Provide one of the following: (1) a comparison of the expected peak pressures and bending moments with the allowable values reported in the WCAP report or (2) justification for other alternate allowable pressure and bending moments with a similar comparison with peak pressures and moments induced in the plant piping.