PGE Portland General Electric Company

September 24, 1980

Trojan Nuclear Plant
Docket 50-344
License NPF-1

Mr. R. H. Engelken, Director
Nuclear Regulatory Commission
Region V
Suite 202, Walnut Creek Plaza
1990 N. California Blvd.
Walnut Creek, CA 94596

Dear Mr. Engelken:

Attached is PGE's response to IE Bulletin 80-18 concerning the potential damaging of the centrifugal charging pumps under low charging flow conditions. Our review has indicated that we cannot assure that this problem will not occur at the Trojan Nuclear Plant, and consequent action is being taken to implement the applicable recommended interim modification. Concurrently, we are examining potential solutions in order to implement a permanent modification at the earliest possible date.

Thus far we have expended approximately 10 man-days reviewing and evaluating the scope of the identified problem, and expect to require a total of approximately 40 man-days to complete design review, implementation, and testing.

If you have any questions, please contact me.

Sincerely,

Donald J. Broehl

c: Mr. Lynn Frank, Director
State of Oregon
Department of Energy

Director
Office of Inspection and Enforcement
Division of Reactor Operations
Inspection
U. S. Nuclear Regulatory Commission

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ATTACHMENT 1

TROJAN NUCLEAR PLANT
RESPONSE TO IE BULLETIN 80-18

NRC Request 1

Perform the calculations outlined in the enclosure for your plant.

PGE Response

The calculations were performed using Trojan plant design data following the procedure outlined in the Bulletin. The result indicates that the centrifugal charging pumps minimum cooling flow may not be assured under the specified conditions.

NRC Request 2

If availability of minimum cooling flow for the CCPs is not assured for all conditions by the calculations in 1:

a. Make modifications to equipment and/or procedures, such as those suggested in the enclosure, to ensure availability of adequate minimum flow under all conditions. If modifications are made as described in the attachment for Interim Modification II, verify that the Volume Control Tank relief valve is operable and will actuate at its design setpoint.

b. Justify that any manual actions necessary to assure adequate minimum flow for any transient or accident requiring SI can and will be accomplished in the time necessary.

c. Verify that any manipulations required (valve opening or closing, along with the instrumentation necessary to indicate need for the action or accomplishment of the action, etc.) can be accomplished without offsite power available.

d. Justify that flow available from the CCPs with the modifications in place will be sufficient to justify continued applicability of any safety related analyses which take credit for flow from the CCPs (LOCA, HELB, etc.).

e. Justify that all Technical Specifications based on the Item 2.d analyses remain valid.

PGE Response

a. The recommended Interim Modification II described in Attachment 1 of the Bulletin will be implemented. This consists of:

   1. Aligning the CCP miniflow line discharge to the Volume Control Tank and isolating the miniflow direct return path to the CCP suction.

2. Removing the safety injection initiation automatic closure signal from the CCP miniflow isolation valves.

3. Modifying the Plant Emergency Operating Procedures to instruct the operator to close the CCP miniflow isolation valves when the RCS pressure drops below 1550 psig, the calculated pressure for manual reactor coolant pump trip, and follow SI termination guidelines if RCS pressure subsequently rises to greater than 2000 psig.

In addition, PGE will conduct the following tests to verify applicability of this problem to Trojan and assure operational requirements for implementing the interim modifications:

1. Verify operating characteristics of the CCPs

2. Perform flow verification tests on the miniflow recirc lines to ensure that the design miniflow rate of 60 gpm is still met for each pump.

3. Perform a pressure test on the Volume Control Tank relief valve to verify its setpoint and operability. The relief valve design flow capacity has been verified to be considerably greater than the combination of CCP miniflow requirements and RCP seal injection return flow.

Implementation of the interim modification will proceed, with completion prior to January 1, 1981. However, testing of the Volume Control Tank relief valve will be deferred until the spring 1981 refueling outage.

Continued operation in the interim before January 1, 1981 is justified by the low probability of pump damage during that period and the acceptability of the consequences following those accidents where Westinghouse predicts pump damage can occur. Westinghouse's analysis methodology conservatively assumes worst-case conditions with respect to pump characteristics, equipment operability, instrument error, relief and safety valve setpoints and system transient behavior. PGE believes that the combined probabilities of a secondary line break and concurrent worst case conditions is acceptably low. Trojan's pressurizer Power Operated Relief Valves are operable and powered from safeguard busses and would be expected to maintain RCS pressure below the safety valve setpoint during a secondary line break. In the unlikely event that both CCPs are damaged following a secondary line break, Plant shutdown and cooldown can be accomplished using the normal charging pump, and if necessary, safety injection pumps.

In addition to implementation of the interim modification, PGE is examining potential permanent modifications to eliminate the need for operator action to isolate or restore CCP recirculation flow.

b. Revisions will be made to Procedures E-0 (Immediate Actions and Diagnostics), E-1 (Loss-of-Coolant Accident), E-2 (Loss of Secondary Coolant), and E-3 (Steam Generator Tube Rupture) to go into effect concurrent with the implementation of the interim modification.

Following implementation of the interim modification, the most limiting conditions requiring operator action occur during small-break Loss-of-Coolant Accidents (LOCAs) in the 2-in. to 6-in.-diameter break size range (see Attachment 2, Review of Accident Analyses), and require operator action to isolate miniflow prior to core uncovery. This occurs approximately 10 min. into the transient, giving the operator adequate time to isolate the miniflow lines.

c. The CCP miniflow isolation valves at Trojan are powered from the 4.16-kV emergency power supply and can be operated with offsite power unavailable. Required instrumentation for monitoring necessary parameters are powered by the preferred instrument 120-V a-c system, a Class 1E system.

d. Attachment 2 is a Westinghouse-supplied generic evaluation of the impact the interim modification has on the safety analyses. If CCP recirculation flow is verified to be consistent with the Westinghouse input assumptions, the Westinghouse evaluation is applicable to Trojan. The conclusion is that the most limiting case, the worst size small-break LOCA, will result in less than a 10°F peak clad temperature penalty if the operator isolates the miniflow recirc lines just prior to core uncovery (approximately 10 min. into the transient). This will not cause the small-break analysis to become more limiting than the large-break LOCA FSAR analysis. The large-break LOCA is not sensitive to the reduction in CCP flow that results from the modifications, and consequently the acceptance criteria presented in 10 CFR 50.46 are still met.

e. PGE has examined the impact of the interim modifications on the Trojan Technical Specifications and concludes that, based on the safety analyses sensitivity studies conducted, the Technical Specifications remain valid.

NRC Request 3

Provide the results of calculations performed under Item 1 and describe any modifications made as a result of Item 2 (include the justifications requested).

PGE Response

Attachment 3 is a summary of our results of the calculations performed under Item 1. Descriptions of the modifications and justifications are included in PGE responses to NRC requests 1 and 2.

Attachment 2

CENTRIFUGAL CHARGING PUMP OPERATION FOLLOWING SECONDARY SIDE HIGH ENERGY LINE RUPTURE

Reference 1: NS-TMA-2245, 5/8/80

Reference 1 notified the NRC of a concern for consequential damage of one or more centrifugal charging pumps (CCP) following a secondary system high energy line rupture. Reference 1 included a calculational method and sample calculation to permit evaluation of this concern on a plant specific basis. Should a plant specific problem be identified, Westinghouse provided several recommendations for the interim until necessary design modifications can be implemented to resolve the problem. These recommendations included two proposed interim modifications which included:

1. Remove the safety injection initiation automatic closure signal from the CCP miniflow isolation valves.

2. Modify plant emergency operating procedures to instruct the operator to:

   a. Close the CCP miniflow isolation valves when the actual RCS pressure drops to the calculated pressure for manual reactor coolant pump trip.

   b. Reopen the CCP miniflow isolation valves should the wide range RCS pressure subsequently rise to greater than 2000 psig.

Prior to making this recommendation, Westinghouse evaluated the impact of the recommended operating procedure modifications on the results of the various accidents which initiate safety injection and are sensitive to CCP flow delivery. The accidents evaluated in detail include secondary system ruptures and the spectrum of small loss of coolant accidents. The analytical results for steam generator tube rupture and large loss of coolant accident are not sensitive to a reduction in CCP flow of the magnitude that results from the recommended modifications. This letter functions to supplement Reference 1 and identify the sensitivity of the accident analyses to the recommended modifications. This evaluation is generic in nature.

Secondary System Rupture

Sensitivity analyses have been performed for secondary high energy line ruptures to evaluate the impact of reduced safety injection flow due to normally open miniflow isolation valves. These analyses indicate an insignificant effect on the plant transient response.

A. Feedline Rupture

Following a feedline rupture, the reactor coolant pressure will reach the pressurizer safety valve setpoint within approximately 100 seconds assuming maximum safeguards with the power-operated relief valves inoperable. With minimum safeguards, the reactor coolant pressure will not reach the pressurizer safety valve setpoint until approximately 300 seconds. The time that the reactor coolant system pressure remains at the pressurizer safety valve setpoint is a function of the auxiliary feedwater flow injected into the non-faulted steam generators and the time at which the operator is assumed to take action. With the miniflow isolation valves open, the peak reactor coolant system pressure and the water discharged via the pressurizer safety valves are insignificantly changed from the FSAR results.

B. Steamline Rupture

The effects of maintaining the miniflow isolation valves in a normally open position was also investigated following a main steamline rupture. For the condition II "credible" steamline rupture, the results of the transient with the miniflow valves open showed that the licensing criterion (no return to criticality after reactor trip) continues to be met. The condition III and IV main steamline ruptures were also reanalyzed assuming the miniflow valves were open. The results of the analysis showed that, even with reduced safety injection flow into the core, no DNB occurred for any rupture.

most plants as determined utilizing the currently approved October 1975 Evaluation Model version, as shown in WCAP-8970-P-A. If miniflow is isolated at the RCP trip setpoint rather than the "S" signal, a reduction in safety injection flow of less than 45 gpm results, averaged for the approximately 50 second period of time separating the two events. This reduction in RCS liquid inventory results in core uncovery less than one second earlier, and has a negligible impact on PCT. If miniflow is isolated at the time of core uncovery, or approximately 10 minutes for break sizes in this range, a greater reduction in RCS liquid inventory results in a core uncovery 10 seconds earlier in the transients resulting in less than a 10°F PCT penalty for the worst size small break. This would not result in any present FSAR small break analysis becoming more limiting than the corresponding large break LOCA FSAR analysis.

If miniflow isolation does not occur at any time into the transient for this category of small LOCA, a PCT penalty of 200°F or more could occur.

C. Small break sizes larger than the worst break through the intermediate break sizes (≥ 6" diameter).

Break sizes in this range have been determined to be non-limiting for small break utilizing the currently approved October 1975 Evaluation Model, WCAP-8970-P-A. If miniflow isolation occurs at the RCP trip time for these break sizes, the negligible effect on PCT presented above also applies. Similarly, if isolation occurs prior to core uncovery, the small (< 10°F) PCT penalty will result as well. However, for these larger break sizes, the time of first core uncovery occurs prior to 10 minutes. If miniflow isolation is not performed until 10 minutes, reduced SI will be delivered during the core uncovery time, which can have a greater impact on PCT. Studies indicate a potential PCT penalty of 40°F resulting for these non-limiting break sizes if miniflow is not isolated until 10 minutes. This is not expected to shift the worst break size to larger breaks, since these breaks are typically hundreds of degrees less than smaller limiting small breaks analyzed with the currently approved Evaluation Model.

Small Loss of Coolant Accidents

Sensitivity analyses have been performed to evaluate the impact of reduced safety injection flow on small break loss of coolant accidents (LOCAs). These analyses indicated that miniflow isolation can be delayed, but it must occur at some time into the small break LOCA transient in order to limit the peak clad temperature (PCT) penalty.

The proposed modification delays miniflow isolation and reduces SI flow delivered by approximately 45 gpm at 1250 psia during the delay time period. The impact of this modification was evaluated based on two isolation times: 1) The time equivalent to the RCP trip time, and 2) approximately 10 minutes in the transient, or just prior to system drain to the break for the worst small break sizes. The second time was evaluated to determine the impact if the operator does not isolate miniflow within the proposed prescribed time. The spectrum of small break sizes are considered to encompass all possible small break scenarios. Only cold leg break locations are considered since they will continue to be limiting in terms of PCT.

A. Very small breaks that do not drain the RCS or uncover the core, and maintain RCS pressure above secondary pressure (< ~2" diameter).

   For these break sizes, it is quite possible that the operator may never isolate the miniflow line, since the pressure setpoint will not be reached, and continued pumped SI degradation will persist. However, this will have no adverse consequences in terms of core uncovery and PCT. No core uncovery will be expected for the degraded SI case, similarly to the base comparison case with full SI. The only effect would be a slightly lower equilibration pressure for a given break size.

B. Small breaks that drain the RCS and result in the maximum cladding temperatures (2" < diameter < 6").

   This range of break sizes represents the worst small break size for

ATTACHMENT 3

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Summary of Results from Calculations Performed
Following the Westinghouse Procedures

Pump P205A Δh @ 60 gpm	5924 ft.
Pump P205A Δh @ 60 gpm	5948 ft.
Total Testing Error	±56.0 psi
Total head loss due to injection piping resistance	140.3 psi
Head loss through the RCS	50.0 psi
Elevation head to be overcome	8.4 psi
Max RCS pressure while maintaining 60 gpm miniflow through weak pump	2308.1 psig
Pressurizer safety valve setpoint plus tolerance and accumulations	2534.7 psig

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