Portland General Electric Company

Bart D. Withers Vice President

July 19, 1985

Trojan Nuclear Plant
Docket 50-344
License NPF-1

Director of Nuclear Reactor Regulation
ATTN: Mr. Edward J. Butcher, Jr., Acting Chief
Operating Reactors Branch No. 3
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington DC 20555

Dear Sir:

PGE Response to Requests for Additional
Information Relating to Generic Letter 83-28

In response to recent letters, additional information concerning Generic Letter 83-28 (Required Actions Based on Generic Implications of Salem ATWS Events) is provided. Attachment A responds to your letter of May 28, 1985 pertaining to Positions 3.1.1, 3.2.1, 3.2.2, and 4.1. Attachment B responds to your letter of June 26, 1985 on Position 1.1. Attachment C provides additional information on Position 4.2.1 requested by your staff during recent conversations.

Sincerely,

BDWithers

Bart D. Withers
Vice President
Nuclear

Attachment

Subscribed and sworn to before me this 19th day of July 1985.

Carole A. Hodgdon
Notary Public of Oregon

My Commission Expires: August 9, 1987

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PGE RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
ON GL 83-28, POSITIONS 3.1.1, 3.2.1, 3.2.2, and 4.1
(NRC LETTER DATED MAY 28, 1985)

NRC Item 1

Position 3.1.1 of Generic Letter 83-28 (GL 83-28) states that licensee reviews should ". . . assure that post-maintenance operability testing of safety-related components in the reactor trip system is required to be conducted and that the testing demonstrates that the equipment is capable of performing its safety function before being returned to service" (underlining added). While your response to this item (PGE letter dated November 4, 1983) states that all safety-related components in the reactor trip system are required to be tested, and references Administrative Order AO-3-9 to support this statement, we note in your response to Position 3.2.1, paragraph a.(2)(c), that the designated individual "Identifies any Periodic Operating Tests (POTs) required to verify that the performance of the equipment/system has been restored and marks the testing block if required. The referenced AO, therefore, appears to provide for occasions when testing will not be required. Please clarify your response so that it may be determined whether or not Trojan conforms to the guidance of Position 3.1.1.

Specifically, please address the following:

a. Please state whether all safety-related components in the reactor trip system will be required to be tested following maintenance.

b. If testing will be waived in some instances, please describe the criteria to be used in granting such waivers.

PGE Response to NRC Item 1 (Position 3.1.1)

a. All safety-related components in the reactor trip system will be required to be tested following maintenance except as described in b. below.

b. Administrative Order (AO) 3-9, "Maintenance Requests", defines post-maintenance testing as testing or checking for proper equipment function after the performance of equipment maintenance. Determination of proper equipment performance may be performed in accordance with an approved permanent procedure; an approved temporary procedure; or an installation check, where the characteristic to be verified can be checked by a qualified craftsman without detailed procedures, detailed acceptance criteria, or technical assistance.

Examples of installation checks as listed in AO-3-9 include:

- Electrical circuits, controls, and relay settings are correct.
- Instrumentation is calibrated and in service as required.
- Limit switches, interlocks, and stops are properly adjusted and set.

The work group supervisor is responsible for ensuring that installation checks in post-maintenance testing are appropriate. The identification of any other surveillance required to verify the performance of the equipment/system is required on the Maintenance Request (MR) per Section IV.B.3 of AO-3-9.

The administrative controls in AO-3-9 will be revised to require a senior reactor operator (SRO) to review MRs and discuss the work performed with the Maintenance personnel to ensure that required testing is adequately specified on the MR. If work is performed outside the scope of the original MR, it will be the responsibility of the Maintenance personnel to review the new scope of work with an SRO to define new testing as appropriate.

NRC Item 2

In your response to Position 3.1.1 of GL 83-28, you also stated that a review of all safety-related tests to ensure that the testing adequately demonstrates that the reactor trip system equipment is capable of performing its safety function before being returned to service would be completed prior to November 15, 1984. In accordance with the request contained in Position 3.1.1, please submit the results of this review.

PGE Response to Item No. 2 (Position 3.1.1)

In conjunction with implementation of the automatic shunt trip modification, a review is currently being performed to verify that testing procedures adequately test the capability of all safety-related components in the Solid State Protection System (SSPS) and reactor trip breakers (RTBs) that are required to operate during a reactor trip. Procedures being reviewed include applicable portions of Periodic Instrumentation and Control Test (PICT) procedures: PICT-10-1, "Reactor Protection System"; PICT-22-1, "Time Response Coordinating Document"; PICT-22-3, "Reactor Trip and ESF Logic Response Time"; and Periodic Operating Test (POT) 12-2, "Emergency Diesel Performance Loss of Offsite Power Diesel Automatic Start and Auxiliary Feedwater Automatic Start". Included in the scope of this review are the proposed testing requirements contained in NRC Generic Letter 85-09, "Technical Specifications for Generic Letter 83-28, Item 4.3". The review of testing requirements will be completed December 31, 1985.

NRC Item 3

As noted above, your response to Position 3.2.1 (particularly the reference to AO-3-9, Section II.B.1) suggests the possibility that some components may not be required to be tested. In addition, however, in contrast to your response to Position 3.1.1, your response to this item does not address the criteria to be used in specifying testing. For example, is the demonstration of the capability of safety-related components to perform required safety functions one of the criteria used in prescribing test requirements. Because of these uncertainties, please also respond for the components covered by Position 3.2.1 to Items a and b listed under question 1, above. In addition, please state whether the prescribed testing will demonstrate the ability of the components to perform their safety functions. If testing will not demonstrate such a capability, please identify and justify the exceptions.

PGE Response to Item 3 (Position 3.2.1)

The response to Item 1, above, describes the post-maintenance testing requirements and guidance applicable to all safety-related equipment. In addition, testing requirements are given in specific operating and testing procedures for certain pieces of safety-related equipment. Examples of specific testing requirements follow:

a. Operating Instruction (OI) 1-9, "Operation of 12.47-kV and 4.16-kV Air Circuit Breakers", and OI-1-11, "Operation of 480-V Air Circuit Breakers", require that whenever a breaker for safety-related equipment is racked-out, then a test start of the component shall be performed after racking back in to verify operability.

b. POT-2-4, "ECCS Boundary and Accumulator Valve Inservice Test", requires that applicable portions of this test be performed prior to returning a subject valve to service following maintenance repair or replacement work on the valve.

It should also be noted that safety-related equipment outage worksheets are generated per AO-3-14 for safety-related equipment whose removal from service is not by a Plant Review Board (PRB) approved procedure which also returns it to service (such as routine PICT, POT, or PET). Item 8 of the safety-related outage worksheet, "Verification Before Returning to Service", contains a sign-off line to confirm that required operability tests have been performed before returning the equipment to service. The completion of the required testing certifies that the accepted means of demonstrating the ability of the components to perform their safety functions has been completed.

NRC Item 4

Position 3.2.2 of GL 83-28 states that licensees should submit the results of their checks of vendor and engineering recommendations to ensure that any appropriate test guidance is included in the test and maintenance procedures or the Technical Specifications, where required. Your response of November 4, 1983, states that the review of Westinghouse Bulletins and Letters not previously in your files would be completed by November 15, 1984. In accordance with the request contained in Position 3.2.2, please submit the results of your check of these documents.

PGE Response to Item 4 (Position 3.2.2)

As a result of the committed review, two Westinghouse bulletins containing test requirements were evaluated in accordance with the Operating Experience Review Program (NUREG-0737, Item I.C.5). Results are as follows:

a. Operational Assessment Review (OAR) 84-31 was issued on February 23, 1984, in response to Westinghouse Bulletin NSD-TB-77-11, dated July 21, 1977, regarding Westinghouse Solid State Protection System (SSPS) Periodic Testing of Safety Injection (SI) Reset Timing Circuitry". The conclusion was that PICT-10-1 already provides for periodic testing of the SI reset circuitry as recommended by the bulletin.

b. OAR 84-32 was issued on February 23, 1984 in response to Westinghouse Bulletin NSD-TB-78-2, dated April 7, 1978, regarding Westinghouse Control Rod Drive Mechanism (CRDM) - Coil Polarity Tests. This OAR concluded that no activities have been performed that might have caused polarities to be revised.

NRC Item 5

Your response to Position 3.2.2 states "Maintenance and test procedures for safety-related equipment were originally developed utilizing equipment technical manuals and vendor correspondence. Thus, vendor and engineering recommendations have been reviewed in the past. A re-review of vendor and engineering recommendations beyond those contained in Westinghouse Bulletins and Letters is not planned." This response appears to be deficient in two areas: (1) it categorically excludes from re-review, all safety-related equipment not covered by Westinghouse Bulletins and Letters; and (2) it does not address whether vendor service bulletins and recommendations received subsequent to the original preparation of the procedures have been incorporated in the appropriate procedures. Please provide technical justification for this position, or

provide a commitment to extend your re-review to Bulletins and Letters applicable to non-Westinghouse supplied safety-related components, and provide a schedule for timely submission of the results of the expanded re-review.

<u>PGE Response to Item 5 (Position 3.2.2)</u>

PGE has been operating Trojan since 1975, which has allowed a significant amount of time for problems with safety-related equipment to become manifest. In addition, feedback of information on problems at other plants has been received from numerous sources, and processed under programs in effect during part, if not all of the time, since the plant became operational. These sources include the following:

a. NRC Information Notices, Circulars, and Bulletins.

b. Vendor Technical Bulletins and Letters.

c. Nuclear Operations and Maintenance Information Service (NOMIS).

d. Nuclear Power Experience Reports.

e. 10 CFR 21 Defect Reports.

f. INPO Significant Event Reports (SERs) and Significant Operating Experience Report (SOERs).

g. INPO Operations and Maintenance Reminders.

PGE's Operating Experience Review Program (OERP) was developed in accordance with NUREG-0737, Item I.C.5, specifically to ensure that technical information, operating experiences (both industry-wide and in-house) and other lessons learned are identified, evaluated, and implemented as necessary. This program has been in effect at Trojan since 1980.

It is PGE's position that past experience and existing programs provide reasonable assurance that significant problems which might have existed have already been identified. The cost and manpower resources required to perform the requested extended review appears to be unwarranted, especially in view of recent utility experiences with such an extended review. However, PGE has established a task force to review all safety-related technical manuals for accuracy and consistency and update the manuals as necessary. This will further ensure that vendor manuals are adequate to support plant operation and maintenance. This review is scheduled to be completed by July of 1987.

NRC Item 6

In your response to Position 4.1 of GL 83-28, you state that the implementation of Westinghouse Bulletin WCD-ELEC-18, of December 17, 1971, would be verified prior to restart from the 1984 refueling outage. Please describe the results of this verification effort.

PGE Response to Item 6 (Position 4.1)

The inspection, performed under MR 83-4516 prior to the restart from the 1984 refueling outage, confirmed the presence of the correct undervoltage trip attachments in both reactor trip breakers, both bypass breakers, and a spare breaker in the Warehouse. No further action is required.

PGE RESPONSE TO REQUEST FOR
ADDITIONAL INFORMATION ON GL 83-28, POSITION 1.1
(NRC LETTER DATED JUNE 26, 1985)

NRC Items

The licensee has stated that past experience and post-trip review of reactor trips have not indicated the need of independent assessment of an event. The licensee has further indicated that if the cause of the trip is unknown, permission from the Duty General Manager is required prior to restart. We find that this action to be taken by the licensee is not sufficient to ensure a plant safe operation. We recommend that if any review guidelines (as stated in Section II.A of this SER) are not met, an independent assessment of the event should be performed by the PORC or a group with a similar authority and experience. However, the licensee has established procedures to ensure that all physical evidence necessary for an independent assessment is preserved.

The licensee has provided for our review a systematic safety assessment program to assess unscheduled reactor trips. We recommend that this program be revised to include a requirement for an independent assessment of the event if any of the review guidelines are not met.

PGE Response (Position 1.1)

Administrative Order AO-3-7, "Post-Trip Review and Permission for Reactor Trip Recovery and Mode Changes", was revised March 12, 1985 to include a requirement for independent assessment of the event if any of the review guidelines are not met. AO-3-7 specifies that in the event of disagreements or unresolved anomalies, the events shall be reviewed by the Operations Supervisor and Manager of Technical Services prior to restart. If anomalies are not resolved, the Plant Review Board must review the event prior to restart.

ADDITIONAL INFORMATION ON GL 83-28, POSITION 4.2.1

Discussions with the NRC staff have indicated that additional information is desired related to Position 4.2.1. PGE has stated in previous responses that periodic maintenance on the reactor trip breakers will be on a yearly basis. Westinghouse Technical Bulletin NSID-TB-83-02 (Revision 1) recommends that maintenance initially be performed on a semi-annual basis. The period might then be extended to 9 or 12 months if experience shows this to be sufficient, providing 200 cycles of the breakers is not exceeded during the time interval.

PGE has always maintained the reactor trip breakers and bypass breakers in accordance with the manufacturer's recommendations, and has experienced no significant difficulties with breaker operation or breaker degradation. Our expected number of breaker cycles per reactor trip breaker per year are as follows:

a.	PICT-10-1, "Reactor Protection System"	24
b.	PICT-22-3, "Reactor Trip and ESF Logic Response Time Testing"	4
c.	Allowance for reactor trips	6
d.	Cycling prior to startup	6
e.	PICT-16-4, "Hot and Cold Rod Drop Time Measurement"	18
f.	PICT-16-2, "Rod Position Indication"	1
g.	MP-1-5, "Reactor Trip and Bypass Breakers"	14
h.	POT-12-2, "Emergency Diesel Performance Loss of Offsite Power, Diesel Automatic Startup, and Auxiliary Feedwater Automatic Starts"	2
i.	Allowance for testing after design changes, special plant tests, etc.	2
	TOTAL	77

Therefore, due to past experience, and the expected number of breaker cycles, PGE considers the maintenance frequency to be consistent with Westinghouse recommendations.

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