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Docket No. 50-344

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

Dear Mr. Withers:

SUBJECT: NUREG 0737, ITEM II.B.2.2, "PLANT SHIELDING" POST IMPLEMENTATION REVIEW.

The staff has completed its review of the above item for the Trojan facility. Based upon the evaluation contained in the enclosed Safety Evaluation Report (SER), we conclude that PGE's post-accident shielding modifications meet staff criteria for NUREG 0737, Item II.B.2.2, and accordingly this item is considered to be closed.

In order to prepare the SER, NRC Region V personnel held discussions with members of your staff at PGE Headquarters and the Trojan facility on February 8 and 9, 1983. We would like to extend our appreciation for the assistance provided by your staff members at those meetings.

Sincerely,

Original signed by:

Charles M. Trammell, III
Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosure:
Safety Evaluation

cc w.enclosure:
See next page

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cc: Michael Malmros, Resident Inspector
U. S. Nuclear Regulatory Commission
Trojan Nuclear Plant
P. O. Box 0
Rainier, Oregon 97048

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

Donald W. Godard, Supervisor
Siting and Regulation
Oregon Department of Energy
Labor and Industries Building
Room 111
Salem, Oregon 97310

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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NRC SAFETY EVALUATION

NUREG-0737 ITEM II.B.2.2 "PLANT SHIELDING"

TROJAN NUCLEAR PLANT

DOCKET NO. 50-344

INTRODUCTION

Following the accident at Three Mile Island, Unit 2, the staff developed the NRC Action Plan, NUREG-0660, to provide a comprehensive and integrated plan to improve safety at power reactors. Specific NUREG-0660 items, approved by the Commission for implementation at power reactors, were issued as NUREG-0737. This Safety Evaluation Report (SER) addresses Portland General Electric's (PGE's) compliance with the recommendations contained in NUREG-0737, Item II.B.2 (Plant Shielding) for the Trojan Nuclear Plant.

DISCUSSION

As discussed in NUREG-0737, Item II.B.2, each licensee was requested to perform a radiation and shielding design review of the spaces around systems that can, as a result of an accident, contain highly radioactive materials. The design review was intended to identify the location of vital areas and equipment where personnel occupancy could be unduly limited, or safety equipment could be unduly degraded, by radiation fields during post-accident operations of these systems. Additionally, each licensee was to provide for safe post-accident access to vital areas through design changes, increased permanent or temporary shielding or post-accident procedural controls. The design review was to determine which types of corrective actions were needed for vital areas throughout the facility. Licensees were to have available the final design details of the implementation of this item for post implementation review by the NRC.

The staff's post-implementation review consisted of 1) an overview of the licensee's shielding design review, 2) an inspection of the shielding modifications made as a result of the shielding design review, and 3) an audit to verify that following the assumed accident plant personnel can leave the control room and safely gain access to selected vital areas.

The following items, associated with NUREG-0737, Item II.B.2, are not evaluated in this SER:

1) Post Accident Sampling System (PASS)

   Shielding for the PASS will be evaluated separately under Item II.B.3 "Post Accident Sampling Capability".

2) Radiation Qualification of Safety Related Equipment

This topic is separately addressed under NRC Multi-Plant Action Item B-60 "Environmental Qualification of Electric Equipment for Nuclear Power Plants".

For the purpose of the shielding design review, those areas which must be accessible to aid in the mitigation of, or recovery from, an accident are classified as "vital areas". This definition of vital areas does not necessarily include all of the vital areas defined in 10 CFR 73.2 for security purposes.

EVALUATION

1. SHIELDING DESIGN REPORT REVIEW

The results of PGE's shielding design review were submitted to the NRC in Reference 1. Discussions between a Region V licensing representative and the licensee were conducted at PGE's Headquarters on February 8, 1983, and an inspection of the shielding modifications was conducted at the facility on February 9, 1983. The purpose of the discussions at the licensee's headquarters was to familiarize the Region V representative with the basis for PGE's shielding design review report. As a result of these discussions, additional information was requested from and supplied by the licensee (Reference 2) to clarify information previously submitted.

Reference 1 describes the criteria and assumptions used by PGE in performing their shielding design review. One assumption made by PGE is that, in a post accident situation, letdown flow through the chemical and volume control system (CVCS) to the volume control tank (VCT) and CVCS holdup tanks might be necessary. Additional shielding was provided to assure that vital areas near the CVCS piping and holdup tanks are accessible in a post-accident situation. At the February 9 on-site inspection, the licensee was requested to provide assurance that, in a post-accident situation, letdown flow would not be released past the VCT and CVCS holdup tanks. In Reference 2, PGE indicated that Functional Restoration Instruction C-1 "Response to Inadequate Core Cooling" will be revised, prior to the end of the current plant outage, to include instructions preventing letdown of reactor coolant in the event of severe core damage. Additionally, the Region V representative observed that existing PGE Emergency Instruction EI-1.1 "SI Termination Following LOCA" contained a caution that primary coolant activity must be verified to be below 700 μCi/gram before letdown can be established. Accordingly, we conclude that adequate assurance that letdown will be confined to the VCT and CVCS holdup tanks in post-accident situations has been provided by the licensee.

Reference 1 also indicated that the reactor coolant system was assumed to remain on natural circulation for at least 7 days prior to activating the residual heat removal system. Based on subsequent evaluation, the licensee determined that this assumption was unnecessary and it was deleted as noted in Reference 2.

2. INSPECTION OF SHIELDING MODIFICATIONS

Several types of potential modifications were identified in Reference 1, including the addition of shielding, reach rods or remote valve operators; and the implementation of procedural changes. In References 2 and 3, the licensee indicated that a detailed evaluation of each potential modification resulted in the following required modifications:

(a) Installation of a valve reach rod for draining the radioactive waste gas surge tank, and installation of steel plate shielding on the radwaste gas surge tank cubicle.

(b) Installation of shielding for the CVCS letdown line and holdup tanks to permit access at the radwaste control panel.

(c) Installation of reach rods for Residual Heat Removal System letdown valves 8734A and 8734B.

(d) Modification to facilitate remote handling of contaminated filters (redesign of grappling lugs).

(e) Relocation of containment atmosphere hydrogen sample lines.

On February 9, 1983 an inspection of the above item (a)-(d) modifications was conducted by the Region V representative. The item (a), (b) and (c) modifications were verified to be completed in accordance with the licensee's shielding design review. (Note: The steel plate shielding for item (a) is shown in Figure 12.3-30 of the Trojan Updated Final Safety Analysis Report (UFSAR). The item (b) shielding for the holdup tank "A" and CVCS letdown line are shown in UFSAR Figures 12.3-31 and 12.3-32, respectively.) The redesign of the grappling lugs on the radwaste cleanup filters (item d) was not inspected, due to inaccessibility. However, the licensee stated that the modification has been completed on all existing filter cartridges, and that the filter cartridge procurement specification was revised to specify the correct grappling lug.

The Region V representative determined that Item (e) was no longer required because a new containment hydrogen monitoring system, that does not utilize the old hydrogen sample lines, was installed. The old sample lines were found to be disconnected and abandoned "in place"

Based on the foregoing, the staff considers the modifications noted above to be complete and acceptable.

3. VERIFICATION OF ACCESSIBILITY TO VITAL AREAS

Access to the following vital areas was verified by traversing the route from the control room to each vital area:

(a) radioactive waste gas surge tank
(b) radwaste control panel
(c) containment hydrogen monitoring system (panels I & II)
(d) RCS sampling system (interim RCS PASS).

No potential sources of radiation (that were not included in the licensee's shielding design review) were identified, and each of the above vital areas was found to be safely accessible.

CONCLUSION

Based on the above considerations, we have concluded that the licensee has acceptably complied with the guidelines of NUREG-0737, Item II.B.2 (Plant Shielding) for the Trojan facility.

REFERENCES

(1) C. Goodwin (PGE) 1-2-80 letter to H. Denton (NRC), License NPF-1
(2) B. Withers (PGE) 3-4-83 letter to D. Eisenhut (NRC), "NUREG-0737, TMI-2 Action Item II.B.2 - Plant Shielding"
(3) B. Withers 1-2-81 letter to D. Eisenhut, License NPF-1