U. S. NUCLEAR REGULATORY COMMISSION

REGION V

Report No. 50-344/90-20

Docket No. 50-344

License No. NPF-1

Licensee: Portland General Electric Company
121 S. W. Salmon Street
Portland, Oregon 97204

Facility Name: Trojan Nuclear Plant

Inspection at: Rainier, Oregon

Inspection Conducted: June 25 - July 20, 1990

Inspectors: M. Miller, Reactor Inspector
F. Gee, Reactor Inspector
T. Scarbrough, NRR
D. Corporandy, Reactor Inspector

Approved By: M. T. Kovack for F.R.H. 8/30/90
F. R. Huey, Chief, Engineering Section — Date

Summary:

Inspection during the period June 25 - July 20, 1990 (Report No. 50-344/88-20)

Areas Inspected: Special unannounced inspection of the licensee's design, engineering, and associated quality verification activities. The inspection was performed by inspectors from the Region V office and from the NRC headquarters office. Inspection procedures 30703, 37701, 64704 and 71707 were used as guidance for the inspection.

Results:

General Conclusions and Specific Findings:

Licensee activities appeared adequate in the areas of Motor Operated Valves (MOV) and Fire Protection with the exceptions of the violations noted below.

Significant Safety Matters:

None

Summary of Violations Identified:

Four violations (1 cited and 3 non-cited) were identified as follows:

1. Cited Violation:

   Vendor notices were not being incorporated into plant records as required by procedure.

2. Non Cited Violations:

   a. An unapproved operating instruction for the K-50 Technical Specification smoke alarm system was found taped to the control room wall near the K-50 panel.

   b. Specific operability controls were not implemented for several items of safe shutdown equipment.

   c. An operability determination for the positive displacement charging pump was performed improperly.

Open Items Summary:

During this inspection, 7 new items were opened; 10 previously identified follow-up items were closed, two remain open.

DETAILS

1. PERSONS CONTACTED

Portland General Electric Company

#*S. Bauer, Branch Manager, Nuclear Regulation
*M. Cooksey, Supervisor, Controls and Electrical Maintenance
#*J. E. Cross, Vice President
#E. Curtis, Procurement Supervisor
#F. dePeratta, Safety Branch Engineer
#*C. Dieterle, Nuclear Plant Engineering Supervisor
R. Fredricksen, Nuclear Plant Engineer
#M. Gandert, Supervising Engineer, Civil
#*M. Hoffman, Manager, ME Branch, Engineering Department
G. Huey, Supervisor, Radiation Protection
#G. B. Jones, NPE Electrical App. R.
G. Kent, Supervisor, Surveillance & Test Engineering
#J. Lentsch, Manager, Personnel Protection
#G. Lian III, Specialist, Fire Protection
#J. Mearns, Supervisor, NPEEB
*T. Meek, Branch Manager, Radiation Protection
*P. Morton, Branch Manager, Plant Systems Engineering
*D. Nordstrom, Compliance Engineer
*C. Olmstead, General Manager
*E. Petersen, Supervisor, Mechanical Maintenance
#R. Reinhart, Unit Supervisor, Instrumentation and Controls
#*J. Russell, Quality Audit Supervisor
#A. Sanchez, Senior Engineer, Fire Protection
*R. Schmitt, Manager, Operations and Maintenance
#*C. K. Seaman, General Manager, NQA
#J. Sepaphur, Mechanical Branch, Fire Protection
*M. Singh, Manager, Plant Modifications
*M. Snook, Quality Support Services Branch Manager
L. Strandorge, Plant System Engineer
D. Swan, Manager, Technical Services
#*D. R. Swanson, Manager, Nuclear Safety Branch
G. Tingley, Plant Systems Control & Electrical Supervisor
*T. Walt, Acting Vice President, Nuclear
*T. Warnick, Plant Modifications Engineering Supervisor
*D. Wheeler, Branch Manager, Quality Inspection
#J. F. Whelan, Manager, Maintenance
J. Wiles, Supervisor, Radiation Protection Planning
D. Williams, Quality Support Group Supervisor
#*W. J. Williams Jr., Regulatory Compliance
P. Yundt, General Manager, Technical Functions

Oregon State Department Of Energy

#*H. Moomey, Resident Inspector
#A. Bless, Resident Inspector

In addition to the personnel listed above, during the course of the inspection the inspectors also contacted other licensee employees, including: operations shift supervisors, health physics and maintenance technicians, engineers, quality assurance staff, and various supervisors.

NRC

*T. Scarbrough, NRR
#*J. Melfi
M. Miller, RV
*F. Gee, RV
*D. Corporandy, RV

*Attended exit interview on June 29, 1990.

#Attended exit interview on July 20, 1990.

2. Motor-Operated Valve (MOV) Design Documentation

The inspectors selected a total sample of 9 MOVs from the Safety Injection (SI), Auxiliary Feedwater (AFW), and Residual Heat Removal (RHR) systems. The sampled MOVs were as follows:

SI MO-8923B, Safety Injection Pump Suction
MO-8835, Safety Injection Pumps to Cold Leg Injection
MO-8802A, Safety Injection Hot Leg Injection

AFW MO-2947A and B, Electrical AFW Pump Discharge to Trains A and B
MO-3170, Turbine AFW Pump Steam Stop Valve

RHR MO-8700A, RHR Pump Suction from RWST
MO-8703, RHR Hot Leg Injection
MO-8809B, RHR to Cold Legs

The inspectors evaluated the design documentation for these MOVs with respect to the size of the motor and operator, the current torque switch setting, and the IE Bulletin 85-03 program. The documents reviewed included the following:

Anchor/Darling Gate Valve Drawing, MO-2947A and B.

Anchor Globe Valve Drawing, MO-3170.

Bechtel Calculation # 25-1, MOV Calculations for IEB 85-03.

PGE Piping & Instrument Diagram (P&ID), Residual Heat Removal System, M-205.

PGE P&ID, Safety Injection System, M-206, Sheet 2.

PGE P&ID, Condensate & Feedwater System, M-213, Sheet 2.

PGE P&ID, Auxiliary Steam System, M-214, Sheet 1.

Westinghouse Owners Group Safety-Related MOV Program Final Report, enclosed with letter dated April 7, 1986, from J.D. Campbell, Westinghouse Electric Corporation.

PGE General Computation Sheet TM-298, dated 7/23/88, MOV Design.

Trojan Nuclear Plant Summary Report of the Safety-Related Motor-Operated Valve Switch Setting Review and Testing Program (December 1987), enclosed with letter, dated December 15, 1987, from D. Cockfield, PGE, to NRC.

Report on Safety-Related Motor-Operated Valve Switch Setting Review and Testing Program for Trojan Nuclear Plant (July 1986), enclosed with letter, dated July 15, 1986, from B. Withers, PGE, to NRC.

Limitorque Data Sheets (1/87), MO-8700A, MO-8703, MO-8802A, MO-8835, MO-2947A and B, MO-3170.

Westinghouse Specification Sheets (2/18/71), MO-8700A, MO-8809B, MO-8802A, MO-8835, MO-8923B.

Gates-Vulcan Gate Valve Assembly Drawings, MO-8700A, MO-8809B, MO-8703, MO-8923B.

Velan Gate Valve Drawing, MO-8802A, MO-8835

Design Basis Document 45A, Auxiliary Feedwater System.

Design Basis Document 49/52 Emergency Core Cooling System.

RDC 86-033, Detailed Construction Package 5, Rev. 0 (12/4/89), Replacement of Unqualified Limit Switches and Torque Switches.

In general, the design documentation was minimal for the sampled MOVs. The documentation was more complete for those MOVs covered by the Bulletin 85-03 program.

The inspectors performed approximate calculations to determine the capability of the following MOV's to perform their design basis functions: MO-8809B, MO-8700A, MO-8703, MO-2947 A&B, MO-3170, MO-8923B, MO-8802A, and MO-8835. Within the limits of accuracy of these calculations, MOV's 8809B, 8700A, 2947 A&B, and 8835 appeared marginal in their capability to provide the required thrust under degraded voltage conditions. The licensee confirmed that these valves are covered within the scope of Trojan's Generic Letter 89-10,"Safety-Related Motor-Operated Valve Testing and Surveillance," action items which include review and documentation of the design basis conditions for the operation of safety-related MOV's.

Where necessary, torque switch settings had been revised for MOV's covered under the Bulletin 85-03 program. Revised torque switch settings were based on actual MOV test data and/or vendor values factored to provide additional margins. The licensee confirmed that documentation of switch settings would also be covered under Trojan's response to Generic

Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance."

The inspectors noted that the licensee is in the process of replacing limit and torque switches, and motors, in a number of safety-related MOVs in an effort to upgrade their environmental qualification. The licensee's program for replacement of limit and torque switches is scheduled to meet their commitment of 6 years, based on the licensee's Justification for Continued Operation (JCO) which provides justification for a 6 year replacement program coinciding with Trojan's 6 year major maintenance schedules for safety related MOV's. The licensee confirmed that environmentally qualified limit and torque switches are being installed as part of the regular valve maintenance. Five years of the six year program are complete.

The licensee mentioned that commercial grade motors were supplied to Limitorque, and that the vendor's quality assurance program with respect to environmental qualification of the motors was under question. The licensee considers the MOVs operable at this time. Justification is provided in JCO 86-05. Currently, the 9 Porter-Peerless DC motors at Trojan have been replaced with DC motors which the licensee has environmentally qualified. The remaining motors are AC, and at present their environmental qualification appears questionable. The inspectors emphasized the need to complete this effort. The inspectors consider that this issue raises the question of a need for a notification in accordance with 10 CFR Part 21 by Limitorque and recommend the referral of this matter to the Vendor Inspection Branch of NRR.

The inspectors also noted that SI MO-8802A had been omitted from the Bulletin 85-03 program in accordance with the licensee's criteria in establishing that program. This MOV, however, will need to be addressed within the licensee's response to Generic Letter 89-10. The licensee confirmed that SI MO-8802A is included in the 89-10 program.

The inspectors provided these findings to the licensee for its attention. As discussed later in this report, the licensee has committed to address weaknesses in design documentation as part of the development of a program in response to Generic Letter 89-10.

3. <u>MOV Walkdown</u>

No MOV maintenance was underway at the time of the inspection, but the inspectors did conduct a walkdown of several MOVs. The licensee removed the limit switch compartment cover of AFW MO-3170 for direct observation by the inspectors. For this MOV, the inspectors noted: the torque switch settings were consistent with licensee documentation, the torque switch limiter plate was installed and limit torque consistent with documentation, the torque switch was manufactured of fibrite, and no grease seepage into the compartment was observed. The inspectors identified no concerns during the walkdown.

4. MOV Diagnostic Equipment

The inspectors reviewed the licensee's use of MOV diagnostic equipment and associated procedures and analysis of MOV data. The inspectors discussed the use of diagnostic equipment with licensee personnel, observed the operation of that equipment in the licensee's training facility, and reviewed Maintenance Procedure MP-12-5.05, Rev. 1 (6/14/90), Motor Operated Valve Diagnostic Testing.

The licensee's diagnostic equipment measures motor current and spring pack displacement for analysis. Licensee procedure MP-12-5.05 includes a check for spring pack relaxation. The licensee currently uses a torque switch bypass value of approximately 10 percent. Torque switch bypass values of 10% have been shown to be inadequate to allow full valve opening/closing at some facilities, however, this is not considered a safety significant issue at Trojan based on Trojan past MOV operating experience. The inspectors noted that as part of the 89-10 documentation effort, the adequacy of the torque switch bypass setting should be evaluated based on maximum thrust during valve stroke information.

5. Completed Maintenance Requests

The inspectors reviewed a list of completed Maintenance Requests (MRs) over the last several years for the sampled MOVs. The MRs reviewed were MR 89-0102, 3-Year Maintenance on MO-8802A, and MR 90-0737, 3-Year Maintenance on MO-8923B.

The inspectors did not identify any concerns with particular MOVs in this area. However, a formal method for ensuring that failure analysis is performed at the maintenance personnel level is not evident. The inspectors concluded that the evaluation of maintenance for generic implications, with respect to the MOV undergoing maintenance and other MOVs, would be improved by providing failure analysis training for maintenance personnel, and by adding a step for such an analysis in the maintenance work documents.

6. Degraded Voltage Considerations

The inspectors reviewed the licensee's consideration of degraded voltage effects to ensure that MOVs will perform their design function. In addition to the design documentation listed earlier, the inspectors reviewed Operational Assessment Review (OAR) 89-24, Rev. 1 (2/2/89), Development of Rated Torque by DC Motor-Operated Valves.

The licensee uses a value of 80% of rated voltage for all degraded voltage calculations. In addition, DC-powered MOVs were more closely scrutinized due to the greater line losses inherent in DC systems. The licensee has acknowledged that consideration of degraded voltage, including cable losses, would be a part of its program in response to Generic Letter 89-10.

7. Inservice Testing Procedures

The inspectors reviewed inservice testing (IST) of the Emergency Core Cooling System (ECCS) and AFW system with respect to valve stroke time. The particular documents reviewed were:

Periodic Operating Test procedure (POT 2-3), Rev. 35 (6/20/90), Safety Injection System ECCS Valves In Service Test.

Periodic Operating Test procedure (POT 5-2), Rev. 18 (6/22/90), Auxiliary Feedwater System Lineups and Inservice Testing.

PGE Memorandum, dated April 12, 1990, NRC Generic Letter 89-04 Valve Stroke Time Review, from G. Swearingen to G. Kent.

The IST procedure for the AFW system specifies the use of "actuation to indicating light" for the measurement of stroke time. The IST procedure for ECCS can be read to imply "light to light," although the licensee asserted that its policy is to use "actuation to indicating light." The wording in this procedure should be revised to clearly identify "actuation to indicating light" for measurement of stroke time. In response to NRC Generic Letter 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," the licensee reviewed the stroke time limits for safety-related MOVs. For the sampled MOVs, the stroke times in the Technical Specifications and test documentation were reported to be less than or equal to the stroke times in the safety analyses. The licensee did identify various inconsistencies to be corrected, such as one of the sampled MOVs not being tested in the direction specified in the Final Safety Analysis Report (FSAR).

The licensee reported that almost all safety-related MOVs have 4-rotor limit switches installed in order to allow setting of the indicating lights at the end of valve stroke. The following is the list of safety-related MOV's which still have 2-rotor limit switches:

MO-3293	MO-3291
MO-3060B	MO-3292
MO-3071	MO-3346
MO-3290	

It was noted that the field change notices for these valves stated that new covers were required to accommodate the 4-rotor limit switches, but that the covers were unavailable; therefore the 4-rotor limit switches were not installed. The inspectors considered this insufficient justification for not committing to install 4-rotor limit switches in these valves. This is an open item to confirm that either the 4-rotor limit switches are installed, or that adequate justification is provided for continued use of 2-rotor limit switches in these valves (90-20-01).

The inspectors questioned the licensee about testing requirements for safety-related MOVs. The licensee assured the inspectors that all safety-related MOVs were tested as required in the applicable test procedures. The inspectors performed a cursory review of Trojan's Technical Specifications in order to confirm the requirement that all safety-related MOVs undergo periodic IST to confirm operability. The inspectors noted cases where automatic power-operated valves were excluded from the testing requirements of Specification 4.0.5. Examples are: Containment Spray automatic valves, Containment System valves, and Component Cooling Water valves (refer to Technical Specification pages 3/4 6-11, 6-13, and 7-13). Since the licensee assured the inspectors that these valves are included in the testing program procedures, the inspectors do not consider this to constitute an immediate safety concern. However, the Technical Specifications and anv related documents should be revised to indicate that power-operated automatic safety-related valves require periodic IST. The inspectors plan to review these documents for these changes in a subsequent inspection (90-20-02).

8. MOV Maintenance Procedures

The inspectors evaluated the licensee's MOV maintenance procedures and the 3-year preventative maintenance procedure. Based on past MOV performance at Trojan, there did not appear to be any safety-related problems with maintenance. The reviewed procedures were:

Maintenance Procedure MP-12-5.01, Rev. 0 (2/20/90), Motor Operated Valves Overhaul of Limitorque Models SMB-000 and SMB/SB-00.

Maintenance Procedure MP-12-5.03, Rev. 0 (2/20/90), Motor Operated Valves Preventive Maintenance Procedure 3-Year Inspection.

Maintenance Procedure MP-12-5.04, Rev. 0 (2/20/90), Motor Operated Valves Switch Inspection, Overhaul, Replacement, and Adjustment for Limitorque size SMB-000 through SMB-5.

The development of these procedures represents a significant effort on the part of the licensee over the last few months. The inspectors emphasized the importance of careful attention to the use of the procedures during the implementation period to ensure that plant personnel understand and follow them properly. It is recommended that formal training in these procedures be provided to MOV maintenance personnel.

Procedures require Beacon 325 grease to lubricate the limit swith gear box. This grease has been found at some other facilities to degrade under high temperature conditions. The licensee intends to continue to use this grease because it is part of the licensee's environmental qualification of MOVs. The inspectors informed the licensee that plant personnel will need to be alert to any problems with Beacon 325, and to be prepared to take necessary action if degradation is observed. Based on the licensee's awareness of this issue, and that Trojan plant operating history has not shown the Beacon 325 grease to degrade, no further action is planned on this issue at this time.

9. Operating Experience Information

The inspectors evaluated the processing and control of operating experience information by the licensee. The inspectors discussed this program with licensee personnel and reviewed Nuclear Division Procedure NPD 100-13, Rev. 2 (12/1/89), Operating Experience Review Program, and Nuclear Safety & Regulation Procedure NSRP 330-2, Rev. 5 (1/5/90), Operating Experience Review Program.

Under the licensee's program, the Operating Experience Review Program (OERP) Coordinator is to review operating information and to prepare, as appropriate, an Operational Assessment Review (OAR) to distribute to licensee personnel. Information notices issued by the NRC are routed to the proper licensee personnel in this manner. The inspectors verified that such an OAR had been prepared for NRC Information Notice 90-40 (June 5, 1990), "Results of NRC-Sponsored Testing of Motor-Operated Valves." Documents developed by the Institute of Nuclear Power Operations (INPO) were also identified in the procedures for the processing of operating experience information. Although INPO Significant Event Reports (SERs) were not specifically mentioned in the procedures, the licensee confirmed that INPO SERs are included.

The inspectors found the plant procedures to be inadequate for the control and processing of vendor information. The inspectors selected Limitorque maintenance updates 88-2, 89-1, and 90-1, and found that the licensee's program had not controlled and processed these Limitorque maintenance updates. Through the responsible actions of one licensee engineer, the Limitorque updates had been evaluated for their affect on plant activities, however, the vendor updates were not processed or maintained as records for future reference. The inspectors determined that the licensee's program places inappropriate reliance on individual engineers to input vendor information for processing. Vendor information omitted from the program might fail to receive the necessary attention in determining its affect on safety-related activities. Further, although specific vendor information might not have an effect on plant activities at the time of receipt, modifications to plant operations or equipment might then cause the vendor information to be important to the safe operation of the facility. This could be particularly significant with respect to new personnel who might be unaware of all relevant vendor information. A problem with the licensee's use of vendor information also occurred in the past with Ruskin fire dampers. Consequently, the inspectors determined that the licensee's program for the control and processing of vendor information appeared to be a violation of Criterion V, in Appendix B, to 10 CFR Part 50 (90-20-3). Following the inspectors' identification of this problem, the licensee reported that action had been taken to input all Limitorque maintenance updates into its operating experience review program. The licensee also reported that it had contacted Limitorque to ensure that all future maintenance updates would be provided directly to the OERP Coordinator for input into the program. The licensee committed to notifying its other major vendors of this arrangement as well.

10. MOV Trending Program

The inspectors evaluated the licensee's efforts to provide for the trending of MOV problems and maintenance work. The inspectors discussed the trending program with licensee personnel, observed the operation of one computer-based trending method, and reviewed the licensee's Maintenance Evaluation and Trending System (METS) Reference Guide (June 1990).

The licensee's program for the trending of MOV problems and maintenance work relies on the use of Corrective Action Reports (CARs) and Maintenance Requests (MRs). The licensee developed the CAR program to replace several previous tracking methods (such as nonconformance reports). The sets of CARs and MRs were said to overlap to a large degree, but the differences in their scope necessitates the trending of both sets of documents. The licensee has developed a computerized method to assist in the trending of information, including MOV problems, provided in the CAR documents. Although the CAR program is new, the licensee has made an effort to input previous MOV documents to allow for trending. Further, the licensee stated that a trending report is prepared every 6 months.

The licensee does not currently include MR's in their documented trending program. In the past, the licensee has relied on a maintenance supervisor to evaluate the MR's for trends. The inspectors did not consider this method of trending to be sufficient. The licensee, however, has under development a Maintenance Evaluation & Trending System (METS) which should be available during Summer 1990. The inspectors reviewed the development package for this system. Following its installation, the licensee will need to establish procedures and conduct training to ensure the proper implementation and use of METS. The inspectors identified several potential trends, such as packing leaks and problems with the manual operation of MOVs, in their review of MRs. Therefore, the inspectors consider the issue of MOV trending to be open for further review during a subsequent inspection. (90-20-04)

11. MOV Training

The inspectors reviewed the documentation for MOV training and discussed that training with licensee personnel. The reviewed documents included:

Training Administrative Procedure TAP-603, Rev. 6 (1/4/90), Technical Staff/Technical Manager Training Procedure.

Self Study Training Module M3-B-01-SG for Motor Operated Valves (Rev. 1, 2/6/90).

On the Job Training Module M3-B-01-OJT for MOVs (Rev. 2, 2/6/90).

Electrician Training Program Qualification Checklist (Rev. 5).

The MOV training focuses on the proper use of written procedures. The licensee reported that plant personnel must complete MOV training before conducting MOV maintenance. The inspectors did not find specific failure analysis training for MOV maintenance personnel, although such training was said to have been provided to more senior plant personnel. Since the licensee's METS program will rely on input from the MR's which are performed by MOV maintenance personnel, the inspectors feel that it would be prudent that MOV maintenance personnel receive training in failure analysis evaluations, or that mechanisms are established to ensure that appropriate review by engineers trained in failure analysis review is implemented. Failure analysis review, including root cause evaluation, is necessary in order to establish an effective trending program. In addition, failure analysis training will provide higher quality "as found" analysis and identification of non-conformances. The inspectors will consider this issue during subsequent inspection of open item 90-20-04, on Trojan's MOV trending program.

12. MOV Thermal Overload Protection

The inspectors evaluated the licensee's selection and setting of MOV thermal overload protection devices. The inspectors reviewed the licensee's Nuclear Plant Engineering Electrical Branch Design Criteria No. 3.3, Criteria for Sizing Thermal Overloads and Circuit Breakers Used in Safety-Related Motor-Operated Valves. The criteria are intended to follow Position C.2 of Regulatory Guide 1.106, in that thermal overload trip setpoints are to be set with all uncertainties resolved in favor of completing the safety-related action. The circuitry alerts the control room operator to a trip of the MOV on thermal overload, by the loss of power to both MOV indicating lights. The inspectors noted instances in the sampled MRs in which undersized heaters had been installed in the thermal overload protection circuitry. As a consequence, MOV motors could trip early, before the valves complete their intended safety functions. The licensee informed the inspectors that the documentation which identified the undersized heaters was revised to include the calculations for determining replacement heater size. The licensee verified that the MRs to replace the undersized heaters with appropriately sized heaters had been completed, and that the JCO for safety-related MOVs with undersized heaters was now closed.

13. Response to Generic Letter 89-10

The inspectors discussed the status of the licensee's response to Generic Letter 89-10 with plant personnel. The inspectors also reviewed a letter dated January 19, 1990, from D. Cockfield, PGE, to NRC, forwarding a response to Generic Letter 89-10, and a draft PGE One Year Response to Generic Letter 89-10 (6/26/90).

In response to Generic Letter 89-10, the licensee is developing a program to test MOVs within the program under design-basis differential pressure and flow conditions where practicable. The licensee tested 3 MOVs under full differential pressure and flow conditions during this outage. The licensee plans to test 30 more MOVs during the next outage, although some might not be tested under full differential pressure conditions. As part of its program, the licensee plans to provide for the periodic verification of MOV switch settings. The inspectors considered the licensee to have made progress in the development of a program addressing the issues stated in Generic Letter 89-10. However, the inspectors consider that the minimal design documentation found for the sampled MOVs increases the importance of the design basis review, the performance of differential pressure and flow testing, and the need to complete these activities in response to the generic letter on a prompt schedule. The inspectors recommended that the licensee review Supplement 1 to Generic Letter 89-10 in developing its program.

14. Outstanding MOV Maintenance

The inspectors considered the licensee's outstanding maintenance work request items and evaluated several MRs in detail. The reviewed documents included:

MR 89-4700, MO-3305A will not isolate.

MR 90-5228, Replace torque switch in MO-8104.

MR 90-5229, Replace torque switch in MO-3295.

MR 88-3138, Replace defective stem nut in MO-8813.

MR 88-3137, Replace defective stem nut in MO-8110.

MR 90-2024, Potential motor shaft key defects in MO-8106.

MR 90-2023, Potential motor shaft key defects in MO-8821A.

MR 90-5097, Spring pack full of grease in MO-8821B.

Table 6.2-1, Containment Isolation Barriers, Trojan Final Safety Analysis Report.

OAR 89-21, dated 10/24/88, Potential defective motor shaft keys in Limitorque motor actuators noted in NRC Information Notice 88-84.

The inspectors did not identify any immediate safety concerns requiring resolution before plant startup. The inspectors did request that the licensee verify that no maintenance items on motor-operated valves, particularly containment isolation valves, existed which may lead to a determination of inoperability, and subsequent entry into an action statement.

The inspectors were concerned with the large number of MRs (59) that remain open, and that some have not been resolved in two years. The continued presence of a large number of open MOV maintenance items could lead to, or be indicative of, a breakdown in the control of MOV maintenance. The licensee should institute a plan to eliminate the backlog of MOV maintenance items on an expedited basis.

15. Rotork MOV

The licensee has one MOV with a Rotork actuator. The Rotork actuator is installed in MO-4005, a discharge isolation valve to the Reactor Coolant Drain Tank. The inspectors reviewed PGE Memorandum, dated November 25, 1986, Rotork Valve Torque Switch Setting, from D. Walters to P. Morton, and CAR C90-325 (6/14/90), EQ Lubrication Requirements for Inboard Containment Isolation Valve MOV-4005. The inspectors determined that plant personnel are aware of the need for separate and special attention to the Rotork MOV apart from its Limitorque MOVs. The inspectors identified no specific concerns in this area.

16. Torque Switch Calibration

The inspectors noted that in the past, several instances had been identified and corrected in which torque switches supplied by Limitorque had not been properly calibrated to spring pack displacement by the vendor. Improperly calibrated torque switches could cause the valve to trip in mid stroke (i.e. not fully open/close), or for the motor not to trip at all, and burnout. The inspectors consider this issue to raise the question of a need for a Part 21 notification by Limitorque, and will refer this matter to the Vendor Inspection Branch of NRR.

17. Interference With Safety Related Valve Handwheel

While reviewing the internal Trojan MOV discrepancy list, the inspector noted that MOV MO-112D to the charging pump was described as having its "... handwheel ... too close to the wall...".

The inspector was concerned because of the operating safety significance and that this plant open item was initiated more than two years ago (5/12/88).

Plant System Engineering (PSE) explained that the adjacent valve had a similar problem and was re-oriented. Re-orienting MO-112D was a more complicated problem, but it was being considered as a possible fix.

The inspector expressed concern because re-orienting MO-112D implied that its existing orientation posed a potential interference. The inspector communicated this concern to the licensee and asked if an evaluation had been performed to assess the potential interference problem.

Evaluation of the orientation of valve MO-112D, and potential interference with the adjacent wall, was completed by the licensee's Nuclear Plant Engineering (NPE) Civil Group on June 29, 1990. The NPE Civil Group concluded that the current valve orientation was acceptable, based on an evaluation which showed that interference of the MO-112D handwheel with the adjacent wall would not occur.

The inspector reviewed the evaluation, and it appeared reasonable.

The inspector emphasized that the plant had been allowed to operate after May 1988, when the potential interference was noted. Between May 1988, and June 1990, the licensee had not performed the necessary evaluations to assure that MOV-112D would be capable of performing its required safety function in the event of an earthquake. This is not considered a violation because of its low safety significance.

Henceforth, when conditions which could potentially impair the operability of safety-related components are identified, the condition should be evaluated. The safety-related component should not be considered operable unless confirmed by the evaluation.

18. MOV Overview

The inspectors found that licensee personnel are personally committed to ensuring the proper performance of MOVs at Trojan. They have assumed responsibility to develop an effective MOV program and maintain a sense of ownership as that program is being developed. This is reflected, in part, by the significant development of MOV procedures over the last few months. Further, licensee personnel appear to understand the basic concerns that led to the issuance of Generic Letter 89-10 and are beginning to take steps to resolve those concerns. Nevertheless, the resolution of the MOV issue applicable to Trojan depends on plant personnel having continued support from licensee management.

19. Safe Shutdown Procedure (64704)

The inspector reviewed the plant procedures to be used in the event of a fire; EFP-0, "Procedure in the Event of a Fire", EFP-1, "Alternative Shutdown for Evacuation of Control", EFP-1.1, "Fire Damage assessment Upon Control Room Evacuation", and EFP-2,"Loss of Service Water". The inspector noted the following concerns:

a. Administrative Controls for Operability of Safe Shutdown Equipment Procedure EFP-0 listed, for each fire area, the equipment which would be available to implement safe shutdown in the event of a fire. For some of the fire areas, there would be only one train or channel of equipment available. In the case of instrumentation, Technical Specifications requires three of four reactor protective channels to be operable. Therefore, administrative controls appeared to allow one of the four channels to be inoperable for an indefinite period of time. For the case where only one instrument would be available, this could result in the only available instrument being inoperable, and thus not meet the requirements for safe shutdown.

The following items of equipment appeared to be the only channel of indication available in the event of a fire in the areas listed.

Item	Fire Areas
Pressurizer Pressure	
PI-405	C1, C14,
PI-403	C2, C4, C6
Pressurizer Level	

LI-461	C1, C2, C14
LI-460	C3, C8
Source Range Flux	
NI-31	C2, C4,
NI-32	C3, C5, C8

Other fire areas also appear to take credit for operability of instrumentation which do not appear to have specific, formal operability controls. The licensee should perform a detailed evaluation of all safe shutdown equipment.

The inspector reviewed the maintenance and operability history of the equipment listed above since January, 1988. The equipment was operable the entire time with the exception of during the outage and, for the source range flux instrumentation NI-32, which was inoperable during seven intervals, the longest of which was 4 days. Based on the above observation, the licensee appears to have adequately maintained the safe shutdown equipment in an operable state without formal operability controls. Therefore this does not appear to be a significant safety concern at this time.

The licensee did not appear to have administrative controls to require that these specific items be operable. The inspector considers these controls necessary to implement the requirements of Appendix R to 10 CFR 50, section L, which requires the capability to safely shut down the plant regardless of the area of the fire. Therefore this appears to be a violation of Technical Specification Section 6.8.1.g., which requires administrative controls be implemented for the fire protection program. Because the particular equipment appears to have been operable this does not appear to have been a significant safety concern, and the violation is considered to be non-cited according to 10 CFR 2 Appendix C, paragraph V.A (90-20-5).

a. <u>Fire Areas M3 and M4</u> The fire areas M3 and M4 are manholes containing cables for safe shutdown equipment for two trains of service water, RHR and CCW. In a Safety Evaluation Report (SER), the NRC granted an exemption for separation of trains based on compensatory actions performed by the licensee. The inspector noted that in the procedure EFP-0, (which is the basic fire response procedure and which lists the actions for fires in most fire areas in the plant), there does not appear to be a reference of the safe shutdown equipment available in the event of a fire in areas M3, M4 or the intake structure. Also, the Attachment A section, (where available equipment is listed for fire areas) does not provide information for M3 and M4. Also, the inspector noted that the initial paragraph of EFP-2 stated that it was the procedure to be implemented in the event of a fire in M3 or M4, but no reference was found in the basic procedure, EFP-0 to state the equipment available in the event of a fire in M3 or M4. It does not appear that an operator would be aware of which equipment would be affected in the event of a fire in M3 or M4. The licensee agreed to list the equipment available in the event a fire initiated in M3 or M4.

b. Fire Area Floor Plans The inspector noted that the fire area floor plans in procedure EFP-0 were several revisions behind the plant controlled drawings. The licensee stated that the floor plans had not changed significantly from the drawings included in the procedure, and that EFP-0 was reviewed according to revision 2 of procedure NPEP-200-1, "Control of Plant Procedures." Section 6.2 requires that, at every scheduled revision, applicability of changes to the plant must be reviewed to determine if changes must be made to the procedure. The inspector noted that, based on the findings of the most recent fire protection audit, the seismic gaps between areas A1, A5, and A6 are not properly represented on the floor plan drawings, since the drawings show three hour barriers, and the gaps have less than a three hour rating. The drawings should be revised. The licensee agreed to review the issue and take appropriate action.

c. Procedure for Operating the Positive Displacement Charging Pump (PDP) EFP-0, Appendix A states that, in the event of a fire in areas A2, A3, A4, and A9, the PDP should be operated to provide charging by referencing procedure EFP-1, Attachment 7. The inspector's review of EFP-1 found no Attachment 7. However, a procedure for local operation of the PDP was listed in EFP-1, Attachment H. Attachment H appears to provide local control of the pump, since PDP cables may run through the affected fire area. The reference to Attachment 7 should be corrected. Also, EFP-0, Steps 3 and 5, tell operators, in the event the "A" Centrifugal Charging Pump (CCP) does not start, start the PDP per, OI-3-5, "Charging, Letdown, and RCP Seal Water". The inspector's concern is that, in the event EFP-0 is implemented, the normal operating procedures for PDP operation should not be used since PDP equipment would be routed through fire areas. In addition, the licensee stated that the PDP is not credited for use in some of the above areas since cooling may not be available. Instead, the "B" CCP is credited since it is deenergized immediately upon entering the procedure. The licensee should review EFP-0 to ensure that the actions are taken to use the appropriate charging pump, since there appeared to be inconsistencies between the procedure and the analysis.

Identification of Equipment in Procedures In the procedure EFP-1, several steps require the operation of breakers to support safe shutdown equipment. The inspector was concerned since some steps appeared to reference the breakers only by breaker number, and did not identify the associated safe shutdown equipment. In the event of a fire and control room evacuation, operators may need to quickly communicate actions and equipment status. Therefore, reference to breaker numbers alone in procedures may make this communication more difficult. The licensee agreed to evaluate the need to reference safe shutdown equipment instead of referencing only breaker numbers.

Insufficient Ventilation for Safe Shutdown Equipment The inspector noted that the procedure required that doors be opened to provide cooling ventilation to some safe shutdown equipment. In reviewing calculations TM-286 and TE-144 which determined cooling requirements, the calculations for panels C137 and C138 concluded that a temperature of less than 103.9 F would be expected. This is less than 0.1 degree F less than the service rating temperature of 104 F. The inspector noted that average outdoor temperature was assumed for this result. In the case of maximum design ambient air temperature, the expected panel temperature was calculated to be 110.4 F. The licensee noted that a 7.7% margin of error was included in the calculation. The licensee agreed to consider additional measures to ensure that the panels can be maintained at a lower temperature. Resolution of this issue is considered an open item (90-20-6).

Operator Routes The inspector noted that the length of time required for operators to travel to areas in the plant to operate safe shutdown equipment was critical in the accomplishment of safe shutdown in the event of a fire in the control room, cable spreading room, Manholes M3 and M4, or the intake structure. The operator routes are not documented in the fire response procedures. Discussions with some operators showed that they appeared to be familiar with the plant layout. Therefore, the inspector did not identify a safety concern. However, the inspector considered it prudent to document the expected routes since there is very little margin in the time required to travel to many of the remote operating stations for some of the safe shutdown equipment. The licensee agreed to review this issue.

20. Operability Determination of Positive Displacement Charging Pump (64704)

The inspector reviewed several operability evaluations for fire protection equipment. One of the evaluations appeared to have been done improperly. The PDP was found to have delivered 97 gpm instead of "greater than 97 gpm" as required by Procedure Periodic Operating Test (POT) 9-4, "Positive Displacement Charging Pump Periodic Test." Step 3.1 of the operability determination stated that the safety function of the item is to provide charging flow to meet Appendix R fire protection requirements. However, the associated engineering evaluation did not appear to include the appropriate requirements. For example, during a safe shutdown, charging volume requirements would include contraction of RCS volume due to cooldown, leakage due to loss of cooling to RCP seals (greater than 21 gpm per seal according to Westinghouse calculation WCAP 10541), and leakage due to spurious pressurizer PORV operation in addition to the normal RCP seal leakage. The evaluation did not appear to address these requirements. Therefore, the licensee should perform this evaluation to address these, and any other safe shutdown charging volume requirements, to address actual PDP charging requirements. This issue does not appear to have high safety significance because the PDP delivered 97 gpm instead of the "greater than 97 gpm." The licensee stated that the basis for the PDP minimum charging requirements included a margin of several gpm.

It appears that the licensee did not implement the requirements of licensee procedure NDP 100-21, step 4.2, which requires a description of how the item's original requirements for the functional capability are met, including fire protection capabilities. Therefore, this appeared to be a violation of Technical Specification 6.8.1, in that procedure NDP 100-21 did not appear to have been implemented. Based on the discussion above, this is considered to be a non-cited violation in accordance with 10 CFR 2, Appendix C, paragraph V.A (90-20-7).

21. Technical Specification Smoke Detector (64704)

The licensee recently installed a computer controlled smoke detector alarm system (K-50) in the control room. This system is the smoke detector monitoring system credited in Technical Specifications. This system also provides status and alarm indication for sprinkler, deluge, and spray systems. Halon systems will be monitored in the future. The licensee stated the system is UL listed and meets NFPA Code. Although this system was declared operable at the time of the inspection, the inspector noted thirteen maintenance orders were outstanding for the K-50 smoke detector. Based on a review of surveillance and maintenance records, and discussions with engineers and technicians, it appeared that many of the surveillance procedures for fire protection systems monitored by the K-50 stated that the smoke detector would alarm during the surveillances. However these alarms did not occur. Also, intermittent alarms occurred and cleared, and the K-50 operating instructions did not always appear to make the K-50 work as expected. (In the case of the spurious alarms, the inspector verified that the K-50 had been declared inoperable). The inspector was concerned that, in some of these cases, the plant was at power, the K-50 was determined operable, and the normal false alarm response had apparently not yet been determined for the system or reflected in the surveillance procedures. The licensee agreed to resolve this concern during the performance of the revision of the surveillance and operating procedures.

Informal Operating Procedure The inspector observed a paper taped to the wall of the control room next to the K-50 smoke detector which appeared to be an informal procedure for operating the K-50 smoke detector. The paper was titled "Operation of the K50", and "Reset K50 as Follows". It listed operations to be performed on the K-50 to clear alarms and reset the system. The inspector was concerned that an informal operating procedure was in use for the Technical Specification smoke detector. At the exit meeting, the licensee stated that the procedure was removed and that only formal operating procedures were in use. This appeared to have been an isolated occurrence. This appears to have been a violation of Technical Specification 6.8.1, which requires formal control of operating procedures. Based on the above discussion, this is considered to be a non-cited violation according to 10 CFR 2, Appendix C, paragraph V.A (90-20-8).

Compliance with UL listing and NFPA Code The inspector reviewed the K-50 design with respect to compliance with the UL listing and NFPA Code requirements. Based on discussions with engineers and review of drawings, the K-50 appears to adequately meet the requirements for supervised digital and integrated detector alarms.

Equipment Operable With Open Panels The inspector observed the back panel of the C-43 smoke detector open for several minutes. The licensee stated that the equipment was considered operable if the purpose of opening the panel is to perform a visual inspection, and no changes to the equipment were anticipated. The inspector noted that changes to the equipment could occur, and that UL standards require equipment to be considered inoperable if a probe over a certain size can be inserted into the opening of the panel. The licensee stated that this requirement would be reviewed and formally documented with respect to the justification for opening fire protection equipment panels and continuing to consider them operable. Based on plant operating history, the inspector did not identify significant safety concerns, this issue will not be followed as an open item.

22. Coordination With Offsite Fire Fighting Resources (64704)

The inspector reviewed the training and qualifications of the Ranier Rural Fire Department, the offsite fire protection agency which would respond to the request for assistance in the event of a fire at the plant.

Training The inspector discussed emergency response and personnel training with fire department employees and fire chief, and reviewed emergency response training records for the fire department personnel. The records indicated that the maximum number of individuals (20 due to security considerations) have participated in annual fire drills in the plant, and several individuals have participated in drills at the plant outside the security area. Records indicated that all paid personnel and most of the volunteers have had site familiarization and radiation protection training, and that this training had been updated periodically. Also, the fire chief and the six paid fire fighters have protected area badges for the plant. It appears that all shifts have fire fighters who will be available to support a request for assistance. In addition, the licensee stated that training had been given to fire departments at St Helens and Clatskanie fire departments in the event additional support was required.

Equipment The inspector observed the fire engines and pumper which would respond to a call, and verified, for a sample of the required response equipment, that the hoses and other required equipment were installed on the trucks, and that the connections appeared to be appropriate for the connections installed at the plant. The inspector observed a Trojan Fire Response Plan dated May 1990 in the cab of the inspected fire engine. Radios had been provided to the fire fighters and appeared to have been labeled with and set to the Trojan emergency frequencies.

No violations of NRC requirements were identified

23. Fire Brigade Training and Qualification (64704)

The inspector reviewed the assignments and qualifications of the three control room shifts and security fire brigade members for the date of July 17, 1990. All shifts appeared to have sufficient number and adequate training to meet Technical Specification and Procedural requirements.

Reportability of the Lack of Licensed Operator as Fire Brigade Leader
CAR C90-5120 dated May 7, 1990 described the lack of a licensed operator as a fire brigade leader on May 6, 1990. The issue was determined to be

reportable only if there were not enough individuals on site to make up a fire brigade, and each of the brigade members must know they are part of the brigade. The inspector considers this may be reportable if the person designated to be the fire brigade leader does not meet fire brigade leader training requirements, or does not know they must respond to a fire as a fire brigade leader. Technical Specification 6.2.2.f, states that a fire brigade of at least 5 members shall be maintained on site at all times, and that the minimum requirements may not be exceeded for more than 2 hours. This item is considered an Unresolved Item (90-20-10).

24. <u>Fire Protection System Surveillance (64704)</u>

The inspector reviewed about 80 records of fire protection surveillances, and surveillances of other systems associated with safe shutdown, which had been conducted over the last year. The surveillances appeared adequate and appeared to have been accomplished within the time intervals required by Technical Specifications. The inspector noted the following concerns:

<u>Transient Combustibles Loading During Outages</u> During the recent outage, records of surveillances of fire extinguishers and transient combustibles documented a large number of transient combustibles and frequent occurrences of moved, blocked or hidden fire extinguishers. These discrepancies appeared to be related to the outage work, and were documented as resolved. However, each subsequent surveillance appeared to document a similar number of new discrepancies. The inspector also reviewed similar findings the local fire department had noted during fire protection inspections during the outage. The licensee's cooperation with the local fire department is encouraging. However, blocked fire extinguishers and transient loading appeared to be a continuing problem during outages. The licensee stated that the individuals temporarily hired for outage work may require increased training in the control of combustible loading and fire extinguisher access, and that General Employee Training would be reviewed and revised as appropriate to increase emphasis on control of transient combustibles during outages.

<u>Quality of Surveillance Procedures</u> As a result of recent licensee initiatives, steps in plant procedures must be performed in the order they appear in the procedure. This is a concern because, in order to adequately perform many of the surveillance procedures, operators must add steps or perform steps out of order. A typical example is, for POT-10-5, "Fire Detection System", during the test of a transfer switch, procedure step 7.19.7 closes a switch to transfer power back to the primary supply. The operator comment states that, in order to accomplish this step, a secondary breaker must be opened and closed. In the past, operators had performed the steps necessary to accomplish the surveillance on their own initiative, and then commented on the surveillance record that the changes should be made to the procedure. Now, as a result of the licensee initiative to perform steps in the order listed in procedures, the operators must stop the surveillance and the licensee must issue a procedure change before continuing with the surveillance.

During the review of records of the 15 fire protection surveillance procedures, the inspector noted 37 operator comments requesting changes to procedure steps to accurately reflect system configuration or allow the procedure to be accomplished successfully. The inspector noted that there already is a large backlog of procedure change requests for the fire protection surveillance procedures. The licensee stated that the schedule to perform surveillances would force the necessary procedure changes to be accomplished. The licensee also stated that, in the event a surveillance could not be performed within the interval required by Technical Specifications, the system would be declared inoperable. Therefore, the schedule to complete the surveillances would force timely revision of the procedures. The inspector considers that, although a methodical review of all requests for a specific procedure would be desirable, expediting changes so surveillances could be performed on schedule appeared to be adequate.

Emergency Battery Lights (EBLs) The licensee has requested NRC approval of the use of EBLs to provide emergency lighting during a safe shutdown. The licensee has implemented the use of EBLs in anticipation of NRC approval. The inspector reviewed the surveillance records for these lights, and noted that the surveillance procedure EDP 5-1.1, "Exide Emergency Battery Light," did not clearly state that the light must be declared inoperable if there appeared to be a need for maintenance. The licensee revised the procedure. The revised procedure stated that if at any time the EBL design performance or reliability is questionable, (e.g., proper float voltage cannot be achieved), declare the unit inoperable and notify the shift supervisor. Based on this revision, the inspector's concern appeared to have been satisfactorily addressed.

Halon Storage Surveillance Procedures Revision 19 of licensee procedure POT 10-4, "Fire Extinguishers and Halon Systems Monthly Inspection," note 7.3.1-2, states that a low pressure reading indicates an inoperable system unless the ambient temperature is low. Step 4.3 states: "Elevated temperatures can cause high pressures in the halon storage bottles." These instructions may be interpreted as evaluation instructions to the technicians and operators performing the surveillances. The inspector informed the licensee that if pressures are outside the acceptance criteria, they should be evaluated by design engineering rather than the operator performing the test.

The inspector reviewed two cases of pressure outside the criteria, and found no associated engineering evaluation, although the conclusions that the bottle pressures were acceptable appeared to be appropriate. During discussions with the inspector, design engineers described formal evaluations which had been done on other occasions to determine acceptability of halon bottle pressure. The conclusions of the operability determinations appeared acceptable, however, the documentation of the basis of the conclusions did not appear complete. The licensee agreed to address this issue.

No violations of NRC requirements were identified

25. Use of Elevators by the Fire Brigade

Revision 1 of the licensee procedure FPP-3, "Fire Brigade Routine Practices", states that the fire brigade may use the elevator to get to the scene of the fire. The licensee stated that the elevator was a three hour rated elevator. The inspector identified the following concerns:

a. The inspector questioned use of the elevators during a fire since the elevators did not appear to be UL listed or FM approved. The inspector's concern is that for elevator control system which are not UL listed or FM approved, the elevator is not protected from stopping at the floor with the fire, and the typical effect of the fire on unapproved elevator control circuits is to call the elevator to the floor with the fire. This could endanger the lives of the fire brigade members. The licensee stated that use of the elevator during a fire would be reevaluated.

b. FPP-3 lists the areas where, if there is a fire, the elevator should not be used. These areas are the cable spreading room, the turbine building or control building switchgear rooms, and the control or auxiliary building elevator power supplies (these rooms were not listed by fire area, although plant procedures list fire areas). The inspector is concerned because this does not appear to list all the fire areas which could potentially affect operation of elevators. Two areas which may fit this description are the switchyard, and areas near the elevator control buttons. Although the elevator is not to be used if there is any doubt as to the safety of its use, the inspector considers that more complete information could provide expedient evaluation of elevator safety and operability.

c. Although fire brigade members are expected to be familiar with the fire area designations and associated equipment, procedures (such as FPP-3) to be used in the event of a fire emergency should be consistent by designating areas either by descriptive room designation (control building switchgear room) or fire area (C4).

d. When requested by the inspector, the licensee could not provide an analysis or drawing of which fire areas through which the elevator power and control cables ran.

The licensee agreed to evaluate the above concerns. Because the action of the fire brigade is not credited for safe shutdown, no violations of NRC requirements were identified.

26. Deviations to the Fire Protection Program

The licensee stated that deviations to the fire protection program were addressed in the following manner:

Administrative Control The licensee stated that the procedure NDP 700-2, "Control of the Trojan Operating License and Licensing Documents", which addresses changes to the fire protection program, requires control to the program in the same manner that changes to the FSAR are controlled. As part of these requirements, the licensee stated that the fire protection program was submitted to the NRC in 1984. Also submitted was a list of exemptions to 10 CFR 50 Appendix R.

Inoperable Technical Specification Fire Barrier The licensee has reported inoperable fire barriers when the requirements of 10 CFR 50.72 or 50.73 are met. Otherwise, the licensee reports inoperable fire barriers according to Technical Specification section 3.7.9, which requires that a report be submitted in accordance with Specification 6.9.2, when a fire barrier penetration is not restored to functional status. The subject fire barriers are the three hour fire barriers and penetrations protecting fire areas which contain safe shutdown or safety related equipment. These barriers were originally approved by the NRC, and documented in the original licensee Fire Protection Program, PGE 10-12, as modified by approved exemption requests or SERs.

The licensee stated that the deviations identified in the 1984 10 CFR 50 Appendix R review to PGE 10-12, were submitted to the NRC as a formal submittal for information.

Also, the licensee performs an evaluation of inoperable fire barriers according to the requirements of Generic Letter (GL) 86-10. The inspector considers the guidance of GL 86-10 to require that these evaluations address the requirements of 10 CFR 50.59. The licensee disagreed, and stated that, although some 86-10 evaluations include 50.59 evaluations, the licensee does not consider a 50.59 evaluation to be a required by GL 86-10. The inspector considers this to be an open item (90-20-9) to be resolved during the Appendix R inspection scheduled for January 1991.

Changes to, and Deviations From, the Fire Protection Program The licensee incorporated the fire protection program as described in PGE 10-12 into the FSAR, in response to the guidance of GL 86-10. Therefore, changes to, and deviations from, the fire protection program are evaluated as changes to the FSAR by performing GL 86-10 and 50.59 evaluations. Specific evaluations are performed for major changes, and annual evaluations are performed for groups of changes and deviations which the licensee considers as minor. The licensee does not consider that exemption requests for all deviations to the fire protection program should be submitted to the NRC. This position is based on the wording of the license condition 2.C.8, which does not specifically require that an exemption request be submitted. The NRCs concern is that, for licensees such as Trojan which have not had fire protection surveillance requirements removed from Technical Specifications by a license amendment, the NRC has required licensees to submit exemption requests for all changes and deviations to the fire protection program to the NRC for review and approval. This requirement is based on the original NRC approval of the licensee's fire protection program. Because the licensee and the NRC are not in agreement concerning the need for exemption requests for all changes and deviations to the fire protection program, NRR will review and determine the basis for the licensee's requirements based on available regulatory information. This, and the concerns discussed below will be resolved by NRR (90-20-9).

Deviations to the NFPA Code The fire protection program states the various NFPA Code requirements with which the licensee fire protection program complies. These requirements are based on the Branch Technical Position 9.5.1, and licensee commitments documented in PGE 10-12. The licensee stated that the deviations to the BTP 9.5.1 invoked Code requirements that are evaluated according to GL 86-10. Deviations to other PGE 10-12 invoked NFPA code requirements are documented and evaluated individually and then evaluated annually as a group as part of the annual fire protection program update. This annual evaluation includes an evaluation in accordance with 50.59. The licensee stated that this process is administratively controlled by procedure NDP 700-2, and that each deviation has been evaluated with respect to the bases and specific requirements of the NFPA code. For example, if a check valve deviates from NFPA code, the evaluation should include the original basis of NFPA approval of check valves in that service, such as brass construction to ensure long term operability, and the manner in which the alternate check valve complies with that basis.

The NRC considers that NFPA Code sections invoked by the fire protection program are part of the fire protection program. Therefore, deviations to those NFPA Code sections are deviations to the fire protection program, and should be individually evaluated and administratively controlled as such. This issue is to be included and resolved with the open item (90-20-9) discussed above.

UL Listing The licensee stated that deviations from UL Listing requirements for fire protection equipment are evaluated in the same manner as deviations to NFPA Code discussed above. Therefore the same NRC concerns apply as noted above. Also, with respect to the evaluation of the deviations to UL Listing requirements, the NRC considers that the UL testing is a necessary qualification of equipment, and any deviations to UL listing requirements should include test data for the alternate equipment to substantiate acceptable equipment performance and show that the alternate equipment meets the same requirements as the UL listed equipment. The licensee does not consider test data to be required. This issue will also be followed in the Open Item (90-20-9) discussed above.

Approved Equipment In some instances, the licensee is required to provide "UL Listed or approved" items. The license considers that the approving agency is not specified, and that the definition of authority having jurisdiction for approvals can be specified by the licensee. The licensee considers that it can provide the approval to satisfy the fire protection program requirement. The NRC considers that, for the purposes of fire protection, the licensee does not have approval authority, and that, in context, "approved" is understood to be by a national fire protection agency, such as UL, Farmers Mutual (FM) NFPA, or American Nuclear Insurers (ANI). This will also be reviewed by the NRC in the resolution of the Open Item (90-20-9) discussed above.

No violations of NRC requirements were identified

27. Walkdown of Fire Protection and Safe Shutdown Systems (64704)

The inspector walked down some of the fire protection systems and fire areas. The systems and barriers inspected appeared to have been maintained according to design basis documents and surveillance requirements.

No violations of NRC requirements were identified

28. Fire Protection Engineering (64704)

System Engineering The inspector discussed the status and functions of the fire protection systems with several of the fire protection system engineers. Based on these discussions, and on walkdowns of the fire protection systems, review of surveillance records and other plant documents, the system engineers appear to be of adequate knowledge and awareness of plant and system requirements.

Design Engineering The inspector discussed several design issues with engineers who had performed evaluations. Although the engineers are not in a specific fire protection group, each engineer appeared to be aware of, and trained in fire protection engineering in their discipline areas. All inspector concerns were resolved except those noted in this report. The inspector reviewed Revision 5 of NPEEB Guideline No. 13, Attachment A, "Checklist for Fire Protection ABP Review", and Revision 2 of NPEP 200-1, Attachment A, "Fire Protection Interface Review Form." They appeared to address an adequate amount of fire protection concerns in all fire protection areas.

No violations of NRC requirements were identified

29. Independent Audits of the Fire Protection Program (64704)

The inspector reviewed the last three annual audits of the Fire Protection Program. The audits were performed by individuals who appeared to meet the specified qualifications listed in the Technical Specifications.

Requirement to Evaluate Fire Barriers The most recent audit observed that the evaluation Civil Fire Protection File C-FP-1.3.9, Seismic Gaps With Non-Rated Seals," had the following deficiencies:

a. The evaluation did not address damage to safe shutdown equipment due to combust[illegible] gasses or fire debris

b. There was no evidence that changes to the Fire Area Matrix and success trees after 1986 had been incorporated, although conclusions of the evaluations are indirectly based on these documents.

c. Portions of the evaluation are based on control of transient combustibles, although AO-10-5 does not control transient combustibles in the vicinity of the non-rated penetrations.

d. Seismic gaps between fire areas A1/A5/A6 are not addressed. These three fire areas should not be considered as separate areas if the gaps are not properly evaluated.

These concerns for fire barrier evaluations were documented as observations and not as findings. It appears that the Quality Assurance (QA) organization considered these items were apparently in conformance with fire protection requirements. The inspector considers the proper evaluation of fire barriers to be part of the commitment to 10 CFR 50, Appendix R, Section G. Therefore, it appears that the criteria used by QA to evaluate fire barriers may not be adequate, and should be reviewed during a future inspection, as well as the validity of this evaluation and other fire barrier evaluations. This issue is considered an Unresolved Item (90-20-11) based on the requirements of Appendix R, Section G, and merits further inspection.

Duration of Audit The fire protection audit issued January 4, 1989 states that the duration of the audit was limited to four days in order to complete the audit within the time requirement of a yearly fire protection audit. Six discrepancies and nine recommendations were made. The licensee appears to have performed an audit of reduced scope to meet schedule. More attention to planning appears to have been given to the audit issued January 10, 1990. In that audit, 38 discrepancies were noted.

Voided Non Conformance Reports (NCRs) The inspector noted that eight NCRs appear to have been voided. The NCRs are 88-307, 391, 398, 402, 408, 411, 428, and 429. These NCRs appear to address deviations to NFPA Code, and are therefore deviations to the fire protection program. Based on the discussion of deviations to the fire protection program above, the NCRs should have been resolved. The satisfactory documentation, corrective action, and resolution of these NCRs is considered an open item which will be verified on a future inspection (90-20-12).

Missing Fire Seal CAR C90-1007 dated March 26, 1990, noted a control room pressure boundary penetration and fire wall penetration with no fire seal. The penetration is for 3/4 inch conduit above door 91. The reportability and evaluation of this unrated seal with respect to deviations to the fire protection program discussed above is considered an open item to be resolved with the open item (90-20-11) discussed above.

Training of QA Auditors The licensee stated that a program to provide intensive training in fire protection to auditors has been implemented. The auditors attend training at Loss Prevention Associates, and will participate in successively increased levels of fire protection training and auditing. The training program appears to have been implemented for areas of expertise outside fire protection, and appears to be an indication of increased licensee commitment to quality assurance.

No violations of NRC requirements were identified

30. Corrective Action for Fire Protection Program (71707)

The inspector noted that there were about 25 outstanding corrective action requests (CARs) applicable to the fire protection program, about 19 of which had corrective action overdue by a few months. In addition, nine of the reports appeared to be compilations of several (about 10) similar NCRs and CARs which had been originated about two years earlier. The inspector was concerned that the corrective action of combining reports and subsequently extending the deadlines for resolution was not providing timely corrective action. Also, many of the reports address NFPA code deviations (and therefore deviations to the fire protection program), which do not appear to have been evaluated according to the requirements discussed above. The apparent inconsistency in the evaluation of deviations to NFPA Code and the fire protection program will be followed in the Open Item (90-20-9) discussed above.

The inspector was informed that the backlog of CARs and NCRs would be evaluated and a schedule to resolve the issues would be implemented.

No violations of NRC requirements were identified

31. General Employee Training (64704)

The inspector noted that G1-C-01-HO, "General Employee Training," page 5, stated "If you encounter a "fire door" blocked open without a _fire patrol' sign attached, close it." The sentences before this statement addressed the need to inform the control room of a breached fire barrier penetration, but they do not obviously inform or require the employee to report the open fire door to the control room. The inspector was concerned that an employee may understand that no report is required for doors, but that they need only be closed. The licensee agreed to change the sentences in the training manual.

The inspector also noted that the description of a fire brigade leader may be inaccurate as written in the General Employee Training. On page 6, it states that the fire brigade leader is normally the assistant shift supervisor but can also be any operator who holds a Reactor Operators license and who has been trained to fight fires. The inspector also noted that the most recent audit of the fire protection program, discussed above, recorded an observation that procedure AO-9-5 does not require any extra training for a fire brigade leader beyond that required for a fire brigade member. The licensee stated that additional training is required for fire brigade leaders beyond standard fire brigade member training. The licensee agreed to review the required training and reflect that additional skills and training for fire brigade leaders are required by revising AO-9-5 and General Employee Training.

No violations of NRC requirements were identified

32. Safety Evaluations (37702)

The inspector reviewed several safety evaluations which had been performed according to 10 CFR 50.59. The evaluations appeared to address appropriate concerns and to provide adequate assurance of plant safety.

In addition, many of the safety reviews appeared to address issues recommended by the Nuclear Safety Analysis Center (NSAC) 125 guidelines. The inspector discussed the incorporation of NSAC 125 guidelines to the licensee's safety evaluation procedure. The licensee stated that many of the NSAC guidelines had been incorporated, and that NSAC 125 guidelines were incorporated in routine training of engineers. The licensee had not made a formal commitment to adopt NSAC 125 guidelines for safety evaluations. However, based on the implementation of many of the NSAC guidelines in the safety evaluation procedure, training of engineers, and safety evaluations which appeared to implement most of the NSAC guidelines, the licensee appears to be performing acceptable safety evaluations. The licensee joined Electric Power Research Institute (EPRI) during the last year.

No violations of NRC requirements were identified

33. Followup of Enforcement Items (92702)

a. (Closed) Enforcement Item 50-344/89-26-01

During an inspection conducted in October 1989, which was reported in RV Inspection Report 50-344/89-26, Emergency Fire Procedure (EFP)-2, "Alternative Shutdown for Complete Loss of Service Water Caused by Fire," Revision 3, was found to be inappropriate to the circumstances in that the procedure did not prescribe:

(1) The locations where dedicated hoses were stored.
(2) The required size of the hoses.
(3) The required length of the hoses.

The procedure was supposed to prescribe specific steps for maintaining a supply of service water to affected safety-related components in the event of the loss of normal service water during a fire.

The inspector verified the licensee's corrective actions by performing a walkdown of the equipment and a review of the related documents. The inspector verified that the required number, size, and length of hoses were in their designated location, Hose House No. 1, and that a sign was posted on the exterior of the door stating that the appropriate fire hoses located on the top shelf were for EFP-2 use only and were not to be removed. EFP-2 was revised by the licensee to include the location and required fire hose length and size in Step 23 of the procedure. Periodic Operating Test (POT) 10-9, "Fire Protection System - Fire Equipment Surveillance," was revised to include the required hoses to the Fire Hose House Inventory List (Pot 10-9-DC). Trojan Nuclear Plant Fire Protection Plan (PGE-1012) was also revised to include the dedicated hose requirements in Section 5.2.5, Outside Hose Stations.

The licensee's corrective actions in response to this violation appeared to be adequate. This item is closed.

b. (Closed) Enforcement Item 50-344/89-26-02

Calculation TNP-83-59, Revision 1, "Fire Pump Flow capability for Appendix R Alternate Cooldown without Service Water System Pumps and Offsite Power," failed to assure the adequacy of the assumption. The assumption was not based on actual measurement. In addition, the assumption was not verified by the person making the calculation nor the calculation reviewer. The connection to Number 1 Fire Hydrant, as it was indicated in the connection configuration in the calculation, was actually greater than 50 feet. In fact, a 100-ft hose would have been used under the conditions to which the calculation applied. At a time much later than that of the calculation, an addition of the sixth connection was made available at the fire pump discharge header manifold.

Temporary Plant Test (TPT)-331 was conducted during the recent 1990 refueling outage to re-verify the results from TPT-251 performed in 1988, and documented items that were omitted from that test such as hose lengths and booster pump flow data. The sixth connection at the fire pump discharge header, which was made available after the calculation, was used for this test instead of the Number 1 Fire Hydrant as specified in the calculation connection configuration. The connections between the fire pump discharge header manifold and the service water manifold were made with six 50-foot 2-1/2" fire hoses.

The inspector verified that Calculation TNP-83-59 was voided as the ability to provide service water supply from the fire pump discharge header was verified by testing.

The licensee's corrective actions in response to this violation appeared to be adequate. This item is closed.

c. (Open) Followup of Licensee Event Report 50-344/90-09 (92700)

Licensee Event Report 50-344/90-09 reported that a discrepancy existed in the setpoint of a function generator module in one of the Engineered Safety Features Actuation System's (ESFAS) functional units. The functional unit setpoint was in excess of the Technical Specification allowable value. The module setpoint was inconsistent with the value specified by Nuclear Plant Engineering Calculations. The module setpoint was higher (less conservative) than the value specified by Nuclear Plant Engineering Calculations and applicable system drawings. The affected ESFAS functional unit served to initiate a safety injection signal when the following conditions existed: high steam flow in 2 of 4 main steam lines coincident with either, low-low average reactor coolant temperature (Tave), or low steam line pressure. The misapplied setpoint affected the high steam flow portion of this ESFAS functional unit's operation. The detail was described in Section 5 of Inspection Report 50-344/90-06.

This event was determined by the licensee as the result of an inadequate process for ensuring that engineering calculations were incorporated into, and accurately reflect in plant calibration settings. In the past, there was no established distribution for Nuclear Plant Engineering Department transmittal of these data. It was left to the initiator to determine appropriate distribution. The licensee committed to generating a formal procedure to ensure that plant instrument settings are reflective of setpoint calculations performed by Nuclear Plant Engineering.

The immediate corrective actions taken by the licensee appeared to be adequate. The affected ESFAS functional unit setpoint was recalibrated. Corrective Action Report (CAR) C90-1005 was specifically generated to resolve the LER issue and was completed. Scaling calculation was redone. Other CARs related to the steam flow instrumentation were generated: CAR C90-3160 for steam flow density compensation, CAR C90-5092 for erratic fluctuation of steam flow at low flow, and CAR C90-5145 benchmarking steam flow differential pressure to feed flow differential pressure. CAR C90-5177 was written to evaluate the concerns within the nuclear instrumentation power range rate circuit. New items added to the Commitment Tracking List (CTL) included: a change in the reactor coolant flow span (drawing inconsistency involving the pressure switch PS-2083/PS-2083A), a change of Tave to the overpressure delta temperature setpoint, and a change of Tave and power percentage to the overtemperature delta temperature setpoint.

In addition, the licensee's documents indicated that the setpoint documents in the E-3 drawing for the other actuations were compared to module settings and that the allowable setpoint band on the instrumentation calibration data sheet (Form I&C-4) matched those on the calculations.

The licensee interim corrective action to control incorporation of calculations into the plant setpoint change process appeared to be adequate. When a nuclear plant engineer completed a calculation that had any potential effect on either instrumentation accuracy or calibration of individual devices, the applicable branch manager was responsible for ensuring that a memorandum was issued from the branch manager of Nuclear Plant Engineering to the branch manager of Plant System Engineering and to the manager of Operations. The memorandum was to include a discussion of the potential changes that the calculation may cause, and a copy of the revised calculation. A distribution list was specified. Plant System Engineering then issued a plant setpoint change if needed. The inspector was given Transmittals TE-200, TE-201, and TE-202 as examples of implementation of this interim control process.

The inspector reviewed a draft copy of Nuclear Division Procedure (NDP) No. 200-15 Setpoint Changes, which was committed by the licensee as a long term corrective action to be implemented by December 1, 1990. The draft copy indicated that the procedure was structured to include a flowchart for a Setpoint Request (Attachment A), definition of responsibilities, and transmittal methods. The guidance for handling chang affecting operability or compliance with Technical Specifications in the draft copy was by referring to Attachment B, Safety Evaluation of NDP 100-5, a Nuclear Safety & Regulation Department checkoff in the Setpoint Change Impact Checklist, and a column in the setpoint change log. The interim progress of the procedure appeared to be adequate.

The licensee appeared to have completed the review of all calibration records relating to other Reactor Trip and Engineered Safety Features Actuation System's instrumentation with the exception of the setpoint for high containment radioactivity (Technical Specification Table 3.3-3, Item 3.b.4). The licensee stated that the setpoints of the containment radioactivity instruments depended on the background existing at the time, and were being adjusted accordingly.

The following items are to be reviewed before the LER can be closed:

(1) Review an approved copy of Nuclear Division Procedure No. 200-15 Setpoint Changes. The licensee committed the procedure to be implemented by December 1, 1990.

(2) Review randomly selected samples of calibration records relating to Reactor Trip and Engineered Safety Features Actuation Systems' instrumentation to ensure the accurate reflection of calculated setpoints in device calibration.

This item remains open.

d. <u>(Closed) Unresolved Item 88-34-04, Potential Operation of Pressurizer PORVs</u>

<u>Background</u> In the event a fire caused spurious actuation of a pressurizer PORV, the licensee determined that fuel damage would not occur if the PORV is closed within five minutes, and normal charging is restored within 40 minutes. Therefore, the licensee's procedure EFP-1 required that the operator open the DC power supply breakers within 5 minutes of determining that a control room or cable spreading room fire occurred.

During an NRC inspection in 1988, the inspection team conducted a walkdown of the procedure for control room evacuation. The team raised the concern that the operator appeared to take longer than 5 minutes to open the PORV DC power supply breakers. This appeared to be inconsistent with sections III.G.3 and III.L.1 of Appendix R to 10 CFR 50.

Discussion The licensee determined that spurious operation of the PORV, as a result of fire induced hot shorts, would be mitigated by the installation of double-pole switches in the PORV main control switch. This would allow the operator to manually deenergize the valve. The licensee stated that two proper polarity shorts would be required to spuriously open the valve subsequent to actuation of this switch. The licensee considers that the probability of this occurring is low, and does not warrant additional design considerations. The licensee stated that the double pole switches discussed above had been installed for the pressurizer PORVs (PCV-456 and PCV-455A), and that the actions in the control room and at the distribution panel would ensure that the pressurizer PORVs would not spuriously open

The inspector reviewed the licensee justification and licensee procedures EFP-1 and EFP-2 which required actions to mitigate spurious PORV operation in the event of a fire. The procedures required deenergization of PORVs, and contained cautions to warn operators of the possibility spurious PORV actuation. The procedure appeared to address the appropriate actions to mitigate spurious PORV actuation in a manner consistent with the licensee's analysis.

Based on the licensee's corrective action, this item is closed.

e. (Closed) Unresolved Item 88-34-10, Spurious Actuation of Motor Operated Valves at High-Low Pressure Interface Boundaries)

Background The inspectors identified a concern that a fire may cause spurious operation of a valve at a high-low pressure interface. This could cause a LOCA because the lower pressure system could not withstand the higher primary coolant pressure.

Discussion

The licensee stated that this concern had been resolved as follows:

For PORVs, the resolution to item d. above, which installed double poled switches. satisfactorily mitigates the possibility of spurious operation.

For reactor head vent valves, spurious operation is not a concern since these valves are closed during normal operation with the fuse for each valve's power removed.

For RHR hot leg suction isolation valves (in series), which are closed in normal operation, the breaker for one of the valves is maintained in the open position, thus preventing spurious operation.

Letdown isolation valves are closed during normal operation and fail closed on loss of electrical power. Spurious operation of these valves is mitigated by opening the appropriate breakers during implementation of EFP-1, at the same time the breakers for the PORVs are opened.

The licensee stated that, based on the above discussion, sufficient protective measures have been established to limit the probability of occurrence of high-low pressure boundary valve spurious operation.

Based on the analysis and the licensee's evaluation, the concern for spurious operation of high-low interface valves appears to have been satisfactorily addressed. Therefore, this item is closed.

f. <u>(Closed) Unresolved Item 88-34-09, Multiple High Impedance Fault Analysis</u>

<u>Background</u> During an inspection in 1988, NRC inspectors noted that there was an apparent lack of resolution of concerns in three areas associated with multiple faults induced by fire; fault analysis, inadequate procedures, and molded-case circuit breakers.

(1) <u>Fault Analysis</u> The licensee had not performed an analysis of the potential effects of high impedance faults (HIF) on a common bus, but had stated that these faults were not likely, and that operator actions would be taken in the event these faults occurred.

The inspector reviewed revision 1 of Impell calculation 0300-087-C001, "Multiple High Impedance Fault Analysis." Its purpose was to evaluate the effects on safe shutdown capability for HIFs occurring on circuits originating from common safe shutdown busses. The calculation identified that safe shutdown bus B01 was susceptible to HIFs, and recommended that in the event of a fire and HIFs on the bus, that operators be instructed to shed non-safe shutdown load R240A (25.87 amps) from the bus. The inspector notes that the calculation identifies that a much higher load of 46.08 amps should be shed in order to mitigate the effects of an HIF on bus B01. The calculation assumption 2.7 states that the HIF contribution to any given circuit is assumed to be 5% of the full load running current. The licensee stated that the probability of all loads on the bus producing HIFs simultaneously is small, and therefore this assumption was conservative.

(2) <u>Inadequate Procedures</u> The licensee had apparently not implemented adequate procedures to address required operator actions during bus restoration from potential damage due to multiple HIFs. The inspectors noted a lack of specific instructions to identify HIFs, and to shed loads from the bus.

The inspector reviewed the procedure EFP-0, "Fire in the Control Room or Cable Spreading Room," and noted a caution statement which warned that bus B01 may be lost without tripping the supply breakers for the faulted circuits. The procedure cautions the operators to re-energize the bus and manually restart only safe shutdown loads as necessary if bus B01 is lost, and multiple HIFs are the suspected cause.

(3) <u>Molded-Case Circuit Breakers</u> The licensee program to verify molded-case circuit breaker operability according to the guidelines

of Generic Letter 81-12 was new. Many of the circuit breakers had not been maintained or tested according to the guidelines. The licensee stated that the above issues had been addressed and resolved in the licensee response to Bulletin 88-10.

Based on the above discussion, the concern for high impedance faults on a common bus appears to have been adequately resolved by the licensee.

g. <u>(Closed) Unresolved Item 88-34-03, Potential Failure of Emergency Diesel Generators (EDGs) Due to Loss of Cooling Water</u>

<u>Background</u> The fire areas in which cables for both trains of the service water system are routed through are the control room. cable spreading room, service water pump room, manholes three and four, and the auxiliary building general area. All other fire areas will have at least one train of service water available for cooling the EDG jacket water. Inspectors were concerned that a fire in these fire areas could cause loss of service water cooling to the EDGs which could damage the EDG within 3 to 5 minutes after an EDG automatic start upon loss of offsite power. The time lines examined indicated that more than three minutes were required for an operator to arrive at the EDGs to shut them down. During a walkdown, the team independently determined that more than three minutes were required to complete these actions.

<u>Discussion</u>

The licensee stated the following specific actions were now incorporated in the alternate shutdown methodology: Prior to control room evacuation: operator action to trip the EDGs, immediately after evacuation: local operator (ACO) actions, including decoupling the control circuits for the service water booster pumps P-148B and P148D, and opening the breakers and decoupling the control circuits to the service water pumps P-108B and P-108C.

The licensee stated that the revised alternative shutdown operator actions to stop the EDGs before evacuating the control room, and manual actions at local control stations to push the emergency stop button, can be completed prior to damage to the EDGs. In addition, the licensee stated that operator actions would be completed at the Train B switchgear room to isolate the service water pump, booster pumps, and the EDGs from the effects of a control room or cable spreading room fire.

The inspector reviewed current revisions of EFP-1, and EFP-2 and noted the steps to ensure that damage to the EDGs is minimized. The inspector noted that the licensee credits nine control room actions prior to evacuation. This is unusually high, since most licensees are credited with only one or two actions prior to control room evacuation. In a Safety Evaluation Report, the NRC stated that the proposed nine control room actions prior to evacuation, appeared to be appropriate for safe shutdown.

The inspector reviewed the time line for the actions to preclude damage to the EDGs. The inspector noted that the time line initiated at the time the decision was made to evacuate the control room. As discussed in earlier NRC inspection reports, a fire could initiate and propagate for several minutes before this decision is made. The licensee stated that the time line analysis starts at the time the decision is made to evacuate the control room. The inspector is concerned that a fire could start in the cable spreading room in the cable tray containing EDG or service water pump control cables. In that case, the operator actions and associated time line analysis may not be valid. This issue will be reviewed in the upcoming Appendix R inspection scheduled for January 1991.

Based on review of the licensee's revised procedure, the licensee's time line analysis of operator actions, and other licensee analysis, and the scheduled NRC review, and duplication of a scheduled item, this item is closed.

h. <u>(Closed) Unresolved Item 88-34-05, Process Variables Temporarily Not Available at Remote Shutdown Station</u>

The inspection team identified that hot leg temperature (Thot), cold leg temperature (Tcold), and source range flux would not be available at the remote shutdown station panel (C-160) for about 40 minutes after an alternate shutdown is initiated.

The licensee stated that the need for reactor coolant system (RCS) temperature and source range flux indication is associated with the restoration of auxiliary feedwater. Source range flux indication is required when a potential exists for increases in reactivity, either by boron dilution or RCS cooldown. Boron dilution will not occur since charging is via the refueling water storage tank, and RCS leakage is isolated for inventory control. RCS temperatures are required to verify natural circulation and monitor RCS cooldown. These functions are required subsequent to AFW flow. Since EDG power is required for AFW flow, the lack of battery backing to ensure continuous readout of these instruments will not impact the requirement to monitor RCS parameters during shutdown. The licensee also stated that a revised time line analysis has shown that power to the Bailey Net-90 system would be restored in about nine minutes. Based on the licensee analysis, and review of the time line issue in the scheduled inspection, this item is closed.

i. <u>(Closed) Open Item 87-34-09, Implementation of Modifications Required by Amendment No. 22 to Operating License</u>

This item addresses that the licensee has not implemented all of the modifications required by Amendment No. 22. During an NRC inspection in early 1990, the licensee stated that some of the required modifications may not have been implemented or had been modified. The licensee stated that the plant configuration was justified based on evaluation. In order to achieve compliance, the licensee submitted an amendment request to NRR on November 30, 1988.

NRR stated that the amendment request was still under review, and has not yet been issued.

Based on licensee evaluation, and NRR review of the modifications and justification, this item is closed.

j. <u>(Closed) Unresolved Item 88-34-02, Requirement to Provide Cooling to Reactor Coolant (RCP) Seals</u>

The inspection team noted inconsistencies between the Westinghouse analysis (WCAP 10541) and the licensee assumption that, during safe shutdown, a loss of seal injection for a period of up to one hour will have no adverse affect on seal injection. WCAP 10541 appeared to document an increasing seal leakage rate of up to 480 gpm per pump.

The licensee stated that, according to revision 2 of WCAP-10541, ten minutes after loss of seal cooling, the RCP leakage rate would increase to a value in excess of 21 gpm per pump for a short period of time, and then rapidly decline to a rate of 21 gpm or less. The increase in leakage rate occurs during the transient heat-up phase and thermal equilibrium phase. The licensee stated that a new analysis was performed using leakage rates of WCAP-10541 and assuming one pressurizer PORV opening for three minutes, an RCP seal leakage rate of 3 gpm per pump for the first ten minutes and an increase to 21 gpm per pump thereafter until seal injection charging is initiated. Based on these assumptions, the licensee stated that RCS makeup would be required within 13 minutes of tripping the charging pumps from the control room. The licensee stated that RCS makeup can be provided within 13 minutes using the Train B centrifugal charging pump (CCP).

The inspector noted that the licensee assumption of 21 gpm per pump does not address the WCAP-10541 leakage rate specified as in "excess of 21 gpm." Also, the licensee has installed double pole switches in the pressurizer PORV control circuits, which would reduce plant leakage, and may allow the licensee to assume the cooling need not be provided as soon in the fire scenario.

The inspector considers that this issue is included in Generic Issue 23, and the actual leakage rates and time required to provide cooling to the RCP seals will be resolved in the resolution of Generic Issue 23. Based on the NRR follow of this issue, this open item is closed to preclude duplicate follow of this issue.

k. <u>(Closed) Open Item 88-34-13, Conformance to National Fire Protection Code</u>

Inspectors noted several items of non-compliance with NFPA code, such as supervision of fire suppression systems, pipe supports for automatic fire suppression system and sizing of fire suppression system piping. In addition, the licensee has identified several other non-conformances, and has justified those non-conformances by analysis.

The inspector reviewed the descriptions of the non-conformances. Based on NRR review, the non-conformances should be reported as deviations to the fire protection program, and ether justified by analysis and 10 CFR 50.59 review, or specifically exempted by NRR as deviations to the program, depending on the state of the license amendment associated with Generic Letter 88-10. This issue is discussed above, and will be resolved as part of the Appendix R inspection. Therefore, since this item is followed under a separate item, it is closed.

1. <u>(Open) Enforcement Item 87-34-01, Qualification of Staff Implementing the Fire Protection Program</u>

Inspection Report 87-34 identified that licensee procedure No NDP 200-1 required a fire protection review of Design Change Packages (DCPs) to assure that fire protection requirements were followed, however, the procedure did not require that a qualified fire protection engineer perform this review prior to or subsequent to the Plant Review Board approval of procedures, changes, or modifications to plant nuclear safety-related structures, systems, or components.

Inspection Report 88-17 identified that the licensee's initial response stated that "...requirements would be equivalent to those provided in BTP C&MEB 9.5-1." However, Inspection Report 89-31 identified that the revised procedure did not conform to the applicable NRC licensing documents. The licensee stated that the Technical Specifications and PGE-8010, "PGE Nuclear Quality Assurance Program" do not explicitly require that a "fire protection engineer" review new designs and modifications. The licensee position is, that various members of the licensee engineering department who are knowledgeable in fire protection system design and the requirements of nuclear plant safety, are qualified to perform there reviews. This item was left open pending NRR review of licensee procedures and training documents.

NRR has not yet made a determination, and therefore this item remains open.

33. <u>Exit Meeting (30703)</u>

The inspectors met with licensee representatives denoted in paragraph 1 on June 29 and July 20, 1990. The scope and findings of the inspection were discussed as described in this report. Licensee representatives acknowledged the inspector's findings.