U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report Nos.	50-272/91-23 50-311/91-23 50-354/91-16
License Nos.	DPR-70 DPR-75 NPF-57
Licensee:	Public Service Electric and Gas Company P.O. Box 236 Hancocks Bridge, New Jersey 08038
Facilities:	Salem Nuclear Generating Station Hope Creek Nuclear Generating Station
Dates:	July 31, 1991 - September 10, 1991
Inspectors:	T. P. Johnson, Senior Resident Inspector S. M. Pindale, Resident Inspector S. T. Barr, Resident Inspector H. K. Lathrop, Resident Inspector A. E. Lopez-Goldberg, Reactor Engineer

Approved: ______________________ 9/26/91
J. R. White, Chief, Projects Section 2A   Date

Inspection Summary:
Inspection 50-272/91-23; 50-311/91-23; 50-354/91-16 on July 31, 1991 - September 10, 1991

Areas Inspected: Resident safety inspection of the following areas: operations, radiological controls, maintenance and surveillance testing, emergency preparedness, security, engineering technical support, safety assessment/quality verification, and licensee event reports.

Results: An executive summary follows.

EXECUTIVE SUMMARY

Salem Inspection Reports 50-272/91-23; 50-311/91-23

Hope Creek Inspection Report 50-354/91-16

July 31, 1991 - September 10, 1991

OPERATIONS (Modules 71707, 92701, 93702)

Salem: The Salem units were operated in a safe manner. Radiation monitoring system actuations were reported, and licensee actions were appropriate. A Unit 1 safeguards equipment cabinet (SEC) failure and associated ESF actuations were appropriately responded to by the licensee. The licensee has plans to replace the SECs, as 28 SEC failures have occurred in the past four years. Following discussion with the NRC, the reporting requirements for the capture of any endangered or threatened sea turtles was satisfactorily modified.

Hope Creek: The Hope Creek unit was operated in a safe manner. Good operator response was observed during a feedwater pump control failure, even though the individuals were only recently qualified in their positions.

RADIOLOGICAL CONTROLS (Modules 71707, 93702)

Salem: Periodic inspector observation of station workers and Radiation Protection personnel implementation of radiological controls and protection program requirements did not identify any deficiencies. The material condition of the post accident sampling system was good. Chemistry technicians were observed as being proficient and effective during sampling and analysis efforts.

Hope Creek: Periodic inspector observation of station workers and Radiation Protection personnel implementation of radiological controls and protection program requirements did not identify any deficiencies. Chemistry, training, and emergency preparedness personnel failed to adequately follow procedures associated with post accident sampling system (PASS) operations. Consequently, deficient conditions involving the operability of the PASS were not documented nor corrected in a timely manner.

MAINTENANCE/SURVEILLANCE (Modules 61726, 62703)

Salem: Routine observations did not identify any deficiencies. A Unit 2 reactor Moderator Temperature Coefficient test was well planned and effectively conducted. A steam generator low pressure protection channel was identified by the licensee to be inoperable for a 26 day period due to personnel error (inadequate self-verification during testing). The licensee remains to determine if the condition was unanalyzed. An engineered safeguards feature actuation occurred during testing of vital bus undervoltage relays.

Hope Creek: Routine observations did not identify any deficiencies. A High Pressure Coolant Injection (HPCI) system actuation occurred during surveillance testing. There was no injection to the reactor vessel. The licensee has not yet determined the root cause of the initiation. After extensive investigation, the licensee was unable to determine a definite root cause of the "D" emergency diesel generator test failure in May, 1991, but has enhanced surveillance procedures in an effort to prevent recurrence.

EMERGENCY PREPAREDNESS (Modules 71707, 93702)

Hurricane preparations by the licensee were proactive and conservative. A Hope Creek emergency drill with full onsite participation accountability appeared to fulfill the drill objectives and provided a meaningful training opportunity.

SECURITY (Modules 71707, 93702)

Routine observation of protected area access and egress showed good control by the licensee.

ENGINEERING/TECHNICAL SUPPORT (Module 71707)

Salem: Review of the management of engineering work activities determined that they were performed in accordance with applicable procedures and properly prioritized and executed. The licensee used prudent engineering practices and a conservative safety approach in the replacement of a reactor coolant system temperature detector and the restoration of the 13 loop cold leg temperature channel. An SEC failure resulted in the initiation of a Unit 1 Technical Specification required shutdown.

Hope Creek: Review of the management of engineering work activities determined that they were performed in accordance with applicable procedures and properly prioritized and executed. The licensee continued the investigation into issues surrounding Filtration, Recirculation and Ventilation System (FRVS) operability. Some degradation of environmentally qualified (EQ) components had occurred and the licensee concluded that the vent fan heaters would not have been able to perform as designed for the full period of performance. A 10CFR21 report was submitted by the heater control panel vendor to the NRC. A reactor building ventilation backdraft isolation damper investigation in September 1990 noted a number of EQ, document and spare parts issues. Licensee actions to promptly address these issues were appropriate.

SAFETY ASSESSMENT/QUALITY VERIFICATION (Modules 40500, 71707, 90712, 90713, 92700)

Salem: Significant Event Response Team (SERT) reports documenting two events which occurred during the last report period were reviewed by the resident staff. Although a weakness was identified in one of the reports, the inspectors concluded that the SERT process had been effectively utilized by the licensee for the assessment of the two events.

Hope Creek: A comprehensive and independent scram review was thorough and effective in identifying common causal factors.

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DETAILS

1. SUMMARY OF OPERATIONS

1.1 Salem Units 1 and 2

Both Salem Units remained at power throughout the report period. As of September 10, 1991, Unit 1 had been on-line for 78 continuous days and Unit 2 for 112.

1.2 Hope Creek

The unit maintained operations throughout the reporting period, with weekly power reductions to support main turbine control valve surveillance testing.

2. OPERATIONS

2.1 Inspection Activities

The inspectors verified that the facilities were operated safely and in conformance with regulatory requirements. Public Service Electric and Gas (PSE&G) Company management control was evaluated by direct observation of activities, tours of the facilities, interviews and discussions with personnel, independent verification of safety system status and Technical Specification compliance, and review of facility records. These inspection activities were conducted in accordance with NRC inspection procedures 71707 and 93702. The inspectors performed normal and back-shift inspections, including deep back-shift (9 hours) inspections as follows:

Unit	Inspection Hours	Dates
Salem	5:00 p.m. - 6:00 p.m.	8/2/91
Salem	5:30 a.m. - 7:30 a.m.	8/19/91
Salem	8:00 a.m. - 12:00 noon	9/7/91
Hope Creek	5:30 a.m. - 7:30 a.m.	8/19/91

2.2 Inspection Findings and Significant Plant Events

2.2.1 Salem

A. Unit 1 Engineered Safeguards Feature (ESF) Actuation

At 7:08 p.m. on August 15, 1991, an ESF actuation occurred when the 1A safeguard equipment cabinet (SEC) spuriously actuated. The SEC starts and stops equipment due to accident and/or loss of power signals from the solid state protection system (SSPS). The partial actuation started the No. 11 safety injection (SI) pump, the No. 11 residual heat removal (RHR) pump and the No. 11 auxiliary feedwater (AFW) pump. Selected containment fans tripped as designed.

Operators were dispatched to the SEC and noted a "MODE OP" light indicating the SEC had actuated. An operator observed that a system failure indicator was also lit. The licensee declared the SEC inoperable, entered Technical Specification (TS) 3.3.2.1 and commenced a Unit 1 shutdown from 100% power. The SEC was reset, and equipment was restored to normal. Six minutes later at 7:14 p.m., a second SEC spurious actuation occurred affecting the same equipment. In addition, the 1A emergency diesel generator (EDG) started but did not load since the vital bus remained powered from offsite. Safety equipment was again returned to normal.

Licensee troubleshooting determined that two circuit boards failed. The faulty chassis that contained these circuit boards was replaced with a spare chassis, and the SEC was tested satisfactorily. The licensee declared the 1A SEC operable, and the unit shutdown was terminated at 30% at 12:14 a.m. on August 16, 1991.

The inspector reviewed the incident report, the troubleshooting surveillance test S1.MD-FT.SEC-0001(Q), control room logs, previous SEC failures, and LER 91-27. The inspector also discussed the event with licensed operators, system engineers and plant management personnel. The inspector determined that licensee response to the event was conservative and appropriate. Subsequent to this SEC failure, another failure occurred on September 5, 1991 (discussed in Section 7.1.B of this report). At the end of this report period, 28 SEC failures had been identified (18 on Unit 1 and 10 on Unit 2) since July 1987. (Previous recent failures were discussed in NRC Inspections 272/91-09, 90-24, 90-22, 90-13, 90-11, 90-04). The inspector noted that the licensee intends to replace the SEC with an upgraded system during the next refueling outage for each unit. The inspector verified that Design Change Package 25C-2267 was scheduled for the upcoming Unit 2 outage beginning in January 1992.

B. Unit 1 Engineered Safeguards Feature (ESF) Actuation - Valve Failure

On August 23, 1991, the Unit 1 steam generator (SG) blowdown valve No. 12GB4 failed closed due to a ruptured diaphragm on its actuator. The valve failed following a stroke test. The licensee characterized this inadvertent valve closure as an ESF actuation (since the valve provides for containment isolation) and properly reported this event to the NRC in accordance with 10CFR50.72. The inspector verified that 12GB4 was subsequently isolated, repaired and satisfactorily retested.

C. Radiation Monitor Engineered Safety Feature (ESF) Actuations

The following ESF actuations occurred and were reported by the licensee during the period:

Unit	Radiation Monitor	Date	Time
2	2R41C	Aug.28,1991	6:03 p.m.
2	2R41C	Sep. 1,1991	1:54 p.m.
2	2R12A	Sep.10,1991	7:15 a.m.
2	2R12B	Sep.10,1991	9:13 p.m.

These events continue to be indicative of the degraded radiation monitor system. Systems responded as designed causing a containment ventilation isolation or a control room ventilation start. As stated in previous LERs and management meetings, licensee actions include short term and long term equipment upgrades. The inspector reviewed licensee actions regarding these events. The licensee intends to submit an LER for these events. No unacceptable conditions were noted.

D. Change in Reporting Requirements For Capture of Endangered Species at Salem

Over the course of the 1991 summer, a large increase in the number of captured endangered or threatened sea turtles occurred at the Salem Nuclear Generating Station. The mechanism for the capture of sea turtles is their impingement upon the circulating water system intake screens. The two types of turtles which have been taken this summer are the loggerhead turtle, a threatened species, and the Kemp's ridley turtle, an endangered species. As of the end of the inspection period, 23 loggerhead turtles had been captured, all but one alive; and one live Kemp's ridley had been captured. In accordance with a National Marine Fisheries Service approved procedure, the licensee holds the captured turtles for a short time to determine their state of health. Subsequently, the turtles are tagged and released at a remote part of the Delaware Bay.

Prior to this year, an average of approximately 3.5 sea turtles per summer had been captured at Artificial Island, with a previous high of ten in 1988. The PSE&G environmental engineering staff has attributed the large increase of captured turtles this year to the especially hot and dry weather, which caused the salt line in the Delaware River to migrate north and produced an abundant food supply for the turtles, thus drawing a larger number of sea turtles to the Artificial Island vicinity.

For each sea turtle taken at Artificial Island, PSE&G is required to notify and provide data on the individual turtle to the National Marine Fisheries Service (NMFS). An informal consultation in accordance with Section 7(a) of the Endangered Species Act was conducted between PSE&G, NRC, NMFS and the Environmental Protection Agency in 1981 to study the impingement of sea turtles at Artificial Island. This informal consultation concluded that operation of the nuclear power plants on Artificial Island would not jeopardize continued existence of these sea turtles and established the requirement for PSE&G environmental licensing to make a report to NMFS for each sea turtle taken at either Salem or Hope Creek. As a result of this NMFS reporting requirement, Salem Station had been reporting each turtle capture to the NRC as a four hour report in accordance with 10CFR50.72(b)(2)(vi), which requires a licensee to report any event "related to the ... protection of the environment, for which ... notification to other government agencies has been or will be made."

Due to the burden placed on the Salem operating crews by the reporting of an unusually high number of turtle captures, PSE&G Licensing discussed the 10CFR50.72 reporting requirement with the NRC. Following discussions between PSE&G, the NRC resident staff, Region I and NRR, it was determined that the individual captures of endangered or threatened sea turtles did not have to be reported in accordance with 10CFR50.72(b)(2)(vi). In the view of the NRC, the intent of this paragraph is to report to the NRC conditions that are directly harmful to the environment (such as inadvertent radiological or chemical releases) for which a press release or off-site notification to other government agencies has been or will be made. Consequently, in August, PSE&G initiated a change to their reporting procedures and ceased reporting turtle captures to the NRC Operations Center. The licensee is still required by a Technical Specification, Appendix B, requirement to inform the NRC resident within 24 hours of a sea turtle capture.

2.2.2 Hope Creek

A. Feedwater Control Failure

On August 3, 1991, with the unit at 100% power, the "C" reactor feed pump (RFP) suddenly increased speed to the high speed stops, causing reactor water level to increase rapidly to the high level alarm setpoint (+40"). At the time, the "A" and "C" RFPs were in automatic control and the "B" RFP was tagged out for maintenance. The nuclear controls operator and the nuclear shift supervisor promptly took manual control of "A" and "C" feed pumps, terminating the level increase at +44". Reactor water level was returned to and maintained at its normal level (+35") by manual control of feed pump speed. A failed dynamic compensator card was found in the "C" RFP control logic. The licensee replaced the card with one from the "B" RFP circuitry, tested operability, and returned feed pump control to automatic within four hours of the transient.

The inspector reviewed this event in detail (including chart recorder traces and annunciator logs) with operations personnel. The inspector concluded that operators had acted promptly and effectively in terminating the transient and manually controlling reactor water level until feed pump control could be returned to automatic. Good support was also noted from Instrument and Control (I&C) technicians and the technical staff system engineer. The inspector noted that these individuals were only recently qualified in their positions.

3. RADIOLOGICAL CONTROLS

3.1 Inspection Activities

PSE&G's conformance with the radiological protection program was verified on a periodic basis. These inspection activities were conducted in accordance with NRC inspection procedures 71707 and 93702.

3.2 Inspection Findings

3.2.1 Salem

A. Post Accident Sampling System (PASS)

The inspector reviewed the Salem PASS, including the administrative controls for system operability. A Technical Specification (TS) interpretation (TSI number ADM-6.8.4.E), date May 8, 1990, requires the PASS to be operable in Modes 1, 2 and 3. A 72 hour TS action statement (TSAS) applies if PASS becomes inoperable. This also requires the licensee to inform the NRC and initiate action to restore the system.

The inspector confirmed the licensee's use of the TSI. The inspector reviewed work order (WO) 910617069 on a leaking PASS valve which required removing the system from servic for less than one day. From discussions with system engineers, operators, chemistry personnel, and from reviewing other WOs, the inspector further confirmed that the licensee considers correcting PASS deficiencies a high priority.

On August 20, 1991, the inspector performed a walkdown of the Salem PASS in the auxiliary building. This walkdown was accompanied by training department personnel. T inspector noted that the PASS material condition was good, and that training personnel we competent and knowledgeable of PASS operation.

B. Chemistry Observations

During surveillance test observations (Section 4.3.1.A), the inspector observed two in-pla chemistry technicians perform reactor coolant sampling and analysis. Procedures SC.CH CA.ZZ-0325(Q), "Boron By Titration", and SC.CH-SA.ZZ-0222(Q), "Sampling Reactor Coolant System and Residual Heat Removal Outlet", were observed. The inspector concluded that chemistry technicians were proficient in their duties, and that the procedu were correctly implemented.

During this review, the inspector noted three material deficiencies associated with the Unit 2 primary sampling cabinet: a handle on a sample valve was broken (missing), a toggle switch was malfunctioning, and a valve position light was not working. None of these deficiencies prevented sampling. However, there were no equipment deficiency tags identifying these problems. The inspector questioned chemistry management personnel regarding these items. The licensee reviewed records and determined that these deficiencies were previously identified and were scheduled for work. The licensee stated that while the tags were not required by procedure, the tags did provide information about system status and would be posted.

3.2.2 Hope Creek

A. Post Accident Sampling System (PASS)

During the week of July 29, 1991, the inspector noted that the liquid sampling portion of the Hope Creek PASS operation was out of service due to flow blockage in the water return line to the torus (through solenoid valves SV643A and B). Consequently, the ability to take residual heat removal and reactor coolant post accident samples was prevented. The inspector reviewed a work order that was initiated to effect repair (WO 910712101) and noted that this operability problem was identified to the Chemistry Department by the Training Department on July 12, 1991, via a written feedback form.

On July 19, 1991, the licensee initiated an incident report (Report No. 91-111) which identified that pipe sealant material had apparently been introduced into the PASS return line while performing containment local leak rate testing during the last refueling outage (January - February 1991). The report noted the post-outage testing on PASS in early March 1991 indicated that the PASS was functioning properly at that time. Subsequently, the licensee completed repairs in accordance with WO 910712101; and the PASS was successfully tested and returned to service on August 5, 1991.

From review of related documents and interviews with chemistry, training, and operations personnel, the inspector learned the following relative to previous PASS operability problems. Attachment 4 provides a summary of the sequence of events.

During an emergency drill exercise on March 15, 1991, the drill observer and two chemistry technicians operating the PASS identified that there was insufficient flow to collect a representative sample. Emergency Preparedness personnel documented these findings but did not report the deficiency to station management and chemistry supervision for resolution until May 17, 1991. Upon receipt of the notification, the Chemistry Department personnel reviewed the reported deficiency and tested the system. At that time the Chemistry Department noted that the PASS appeared to be functioning properly and took no further action.

Concurrently, from the period between May 16 and July 9, 1991, the Training Department conducted training of chemistry technicians on the PASS. Several times during this period, some training instructors and chemistry technicians identified intermittent flow problems, such that representative samples could not be reliably obtained from use of the PASS. Reportedly, these problems were identified to the Chemistry Department several times but never documented until July 12, 1991.

Technical Specification (TS) 6.8.1 requires that procedures be established, implemented and maintained covering the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, and the procedures required to implement the requirements of NUREG-0737. Further, TS 6.8.4.C requires the establishment, implementation, and maintenance of a program to include procedures for post-accident sampling and analysis, including provision for the maintenance of sampling and analysis equipment.

Accordingly, the licensee established Procedure HC.CH-EO.SH-0001(Q), "Post Accident Sample Panel Operation." Section 2.9 of that procedure requires the PASS sample team to immediately inform chemistry supervision when any problems encountered during sampling, in order to effect resolution. The inspector noted that the sample team's failure to inform chemistry supervision, until May 17, 1991, of the inability to obtain a representative sample due to flow problems on March 15, 1991, constituted an example of a violation of TS 6.8.1.

Procedure NC.NA-AP.ZZ-0009(Q), "Work Control Process," Revision 2, was also established in accordance with TS 6.8.1. Sections 3.1 and 5.2, requires personnel to initiate work requests and recommend the hanging of Equipment Malfunction Information System (EMIS) tags for malfunctioning components or systems. The inspector noted that failure of licensee personnel to document and initiate work requests for the frequent and intermittent PASS flow problems that prevented representative sample acquisition, and to recommend the posting EMIS tags on the equipment, for the period between May 16 and July 9, 1991, constituted a second example of violation of TS 6.8.1. (50-354/91-16-01)

The inspector noted that the licensee's regard for the importance of maintaining the PASS operable was inconsistent relative to the attention afforded the Salem PASS (See Section 3.2.1.A). For example, the licensee had not established a Technical Specification Interpretation for TS 6.8.4. relative to the expected operability requirements for the Hope Creek PASS. As a result, a lower consideration has been applied to the maintenance and operability of the Hope Creek PASS. Consequently, even after WO 910712101 was initiated on July 12, the system remained out of service until August 5, 1991, since repair was considered as a low priority.

As a result of an independent assessment of this matter by the plant's Quality Assurance Department, the licensee has initiated action to direct more management attention oversight to the operability and maintenance of the Hope Creek PASS, including the identification of root causes and more immediate corrective actions for identified deficiencies.

4. MAINTENANCE/SURVEILLANCE TESTING

4.1 Maintenance Inspection Activity

The inspectors observed selected maintenance activities on safety-related equipment to ascertain that these activities were conducted in accordance with approved procedures, Technical Specifications, and appropriate industrial codes and standards. These inspections were conducted in accordance with NRC inspection procedure 62703.

Portions of the following activities were observed by the inspector:

Unit	Work Request (WR)/Order (WO) or Procedure	Description
Salem 1	Troubleshooting Plan	1A Safeguards Equipment Cabinet (SEC)
Salem 1	Troubleshooting Plan	Loop 13 cold leg temperature instrument
Salem 1	WO910617069	Post Accident Sampling System (PASS)
Salem 1	WO910905104	Reactor Trip Breaker "A" Replacement
Hope Creek	WO910712101	PASS
Hope Creek	WO910819083	PASS
Hope Creek	Various	"B" Reactor Feed Pump shaft seizure investigation and repair
Hope Creek	Various	Rosemount transmitter replacements

The maintenance activities inspected were effective with respect to meeting the safety objectives of the maintenance program.

4.2 Surveillance Testing Inspection Activity

The inspectors performed detailed technical procedure reviews, witnessed in-progress surveillance testing, and reviewed completed surveillance packages. The inspectors verified that the surveillance tests were performed in accordance with Technical Specifications, approved procedures, and NRC regulations. These inspection activities were conducted in accordance with NRC inspection procedure 61726.

The following surveillance tests were reviewed, with portions witnessed by the inspector:

Unit	Procedure No.	Test
Salem 1	S1.MD-FT.SEC-0003(Q)	1C SEC
Salem 1	1IC18.1.013	Reactor Trip Breaker Operability
Salem 2	Reactor Engineering Manual - Part 9	Moderator Temperature Coefficient
Hope Creek	OP-ST.GK-001	"B" Control Room Emergency Filter Monthly Surveillance

The surveillance testing activities inspected were effective with respect to meeting the safety objectives of the surveillance testing program.

4.3 Inspection Findings

4.3.1 Salem

A. Steam Generator Pressure Protection Channel Inoperable Due to Personnel Error

On August 7, 1991, a technician identified that the two lead-lag controller switches were in the test position for the Unit 1 No. 13 steam generator (SG) pressure channel III, rendering that channel inoperable. This condition was identified during the performance of the SG pressure channel III surveillance test. The licensee determined that the two test switches were inadvertently left in that position during the previous channel functional test on July 12, 1991. Upon discovery, the lead-lag controller switches were returned to normal position, thereby restoring the channel to an operable status. Since the Technical Specification (TS) operability requirements were not satisfied (inoperable channel to be placed in tripped condition) from July 12 - August 7, 1991, the licensee reported this event to the NRC in accordance with 10CFR50.73 reporting requirements (30-day licensee event report - LER No. 91-26).

The licensee determined that the root cause of this event was personnel error due to inadequate self-verification of the technician performing the surveillance test on July 12, 1991. The event will be reviewed with applicable personnel. The licensee stated that in addition, the functional surveillance test procedure will be revised to require an independent verification of the lead-lag controller test switch position restoration. The inspector noted that an independent verification should have already been part of the test procedure, as the test is performed on a safety related system. The licensee stated that independent verification is currently required for such procedure steps, however, the test procedure for this event was developed prior to the current requirement and had not yet been revised. The licensee initiated action to review similar procedures to assure that independent verification checks are accomplished.

There is one SG main steamline pressure monitor for each of the four SGs, which provide input to several safety-related circuits, including safety injection (SI). The four main steamline pressure signals are divided into two protection sets; Protection Channel III (Nos. 12 and 13 pressure channels), and Protection Channel IV (Nos. 11 and 14 pressure channels). The affected SI signal is high steam flow coincident with either low-low average reactor coolant temperature or low steamline pressure. Any two of the four low steamline pressure signals will satisfy the low steamline pressure trip coincidence. The purpose of the lead-lag controller is to amplify the incoming steamline pressure signal such that the SI is initiated before the actual steamline pressure reaches the trip setpoint value. Accordingly, the lead-lag controller is credited in the accident analysis.

The licensee's analysis of the above condition identified that a potentially unanalyzed condition existed. A failure of Protection Channel IV (single failure) would result in the delay of pressure channel Nos. 11 and 14 to provide the safeguard actuation signals necessary if called upon during a small steamline break. Under this condition, coincident with the inoperable pressure channel No. 13, the required SI would be delayed due to the mispositioned lead-lag controller switches. The licensee stated that only the small steamline break accident was of concern, since the lead-lag function is not pertinent for larger steamline breaks involving large and immediate steamline pressure drops. This concern was conservatively reported to the NRC upon discovery on September 6, 1991 in accordance with 10CFR50.72 reporting requirements (unanalyzed condition). The licensee is currently reviewing this issue to determine whether the condition is bounded by existing accident analyses.

The inspector reviewed LER No. 91-26 and found it to be acceptable. However, the report references a continuing review of this matter to determine its safety significance, but does nc indicate that a supplemental LER will be provided. The inspector discussed this concern with the licensee, who stated that a supplemental LER will be submitted upon completion of their review. The inspector had no further questions at this time.

B. Unit 2 Moderator Temperature Coefficient (MTC) Measurement

On August 21, 1991, the inspector observed implementation of a surveillance test "MTC Measurement" on Unit 2. The test was required per Technical Specification (TS) 4.1.1.3.b to ensure the value of MTC meets TS requirements when 300 ppm critical Boron concentration is achieved.

The test involved maintenance, operations, reactor engineering and chemistry personnel, ar was performed in accordance with the procedure, Reactor Engineering Manual (REM)-Par 9. The inspector observed test activities from the control room and in the chemistry lab (s section 3.2.1.B). The inspector concluded that the test was well planned and conducted. Personnel performance was commendable.

During test procedure review, the inspector noted that procedure REM-Part 9 does not follow the surveillance procedure format as required by Administrative Procedures NC.NA-AP.ZZ-0032 and AP-12. In particular, acceptance criteria were not included in the procedure steps. However, it was included in an attachment to the procedure. The inspector discussed this with licensee personnel. Their response was that the REM procedures were being currently revised by both the reactor engineering section and by the Procedure Upgrade Project to meet administrative procedure format requirements. The inspector reviewed the licensee's schedule to upgrade REM procedures and concluded it to be acceptable.

C. Unit 2 Engineered Safeguard Feature (ESF) Actuation During Surveillance Testing

On August 26, 1991, an ESF actuation occurred while operating at 100% power when the 2A safeguards equipment cabinet (SEC) was inadvertently actuated during surveillance testing. While performing test procedure No. S2.MD-FT.4KV-0001(Q), "ESFAS Instrumentation Monthly Functional Test - 2A 4kV Vital Bus Under Voltage" a technician applied an electrical jumper across contacts in the wrong relay. This action actuated the 2A SEC, which automatically completed an electrical load shed on the 2A 4kV vital bus, started the No. 2A emergency diesel generator (EDG), and sequentially started associated safety related components. All systems functioned as designed. Control room operators entered procedure No. AOP-ELEC-4kV-A and verified the automatic actions. The 2A vital bus was subsequently restored to a normal lineup. The 2A EDG was subsequently secured and returned to a standby status. Operation of the unit was unaffected by this event.

The inspector reviewed this event and determined that the cause was similar to a previous ESF actuation that occurred on June 6, 1991, discussed in NRC Inspection Reports 272 & 311/91-15 and 272 & 311/91-19. The cause of the June 6, 1991 event was determined to be personnel error caused by human engineering deficiencies. Specifically, technicians are required to install an electrical jumper across contacts on the underside of relays, which are located on the inside of 4kV vital bus cubicle doors, and are positioned approximately nine inches from the floor. Adjacent relays are located about 1/2 inches apart. On June 6, 1991, the technician accidentally touched an adjacent relay while approaching the relay to be jumpered. On August 26, 1991, the technician properly located and identified the proper relay while standing up (the label is above the relay). However, after he positioned himself on the floor to install the jumper, the technician inadvertently connected the jumper to the adjacent relay.

As a result of this latest event, Plant Operations requested that further undervoltage relay testing be suspended until an appropriate hardware change is implemented to prevent further occurrences. There are three 4kV vital buses per unit, and due to equipment concerns (NRC Unresolved Item 311/91-05-01), the undervoltage testing is being conducted on a weekly frequency. The subsequent licensee actions included (1) installing color coded test jacks to the jumper connection points so that the jumpers can be readily installed and removed, (2) providing additional relay labelling to the underside of the relays, and (3) changing the associated test procedure to reflect the above test jack implementation and use. The inspector verified the implementation of the above changes and did not identify any deficiencies.

The inspector noted that the licensee's previous corrective actions were not timely when considering the existing high testing frequency. However, the actions implemented following this event appear to be effective in preventing future similar occurrences. Longer term corrective actions, as stated in previous licensee event reports and NRC inspection reports are also planned by the licensee. The inspector had no further questions.

4.3.2 Hope Creek

A. High Pressure Coolant Injection (HPCI) Initiation During Surveillance Testing

On August 15, 1991, technicians were performing a drywell pressure (B21-N694A) channel calibration. The procedure required the installation and use of a test device which simulated the operation of a number of relay contacts in the logic circuitry. After receiving the fourth drywell high pressure alarm (as expected), the nuclear controls operator (RO licensed) noticed the HPCI steam admission valve HV-F001 stroking open. After verifying that other plant parameters were normal, the operator judged the initiation signal to be spurious and tripped the HPCI turbine before injection to the reactor vessel occurred. The HPCI system was then returned to its standby configuration.

The licensee's immediate investigation did not reveal the cause of the spurious initiation signal. The test was rerun using a different test device with satisfactory results and no unexpected actuations. The licensee also determined that personnel error had not been involved.

The inspector verified that the test device is included in the licensee's measurement and test equipment (M&TE) program, although no periodic calibration of the device is required. Extensive bench testing of the test device did not reveal any malfunctions.

As of the close of this reporting period, the licensee's investigation was ongoing. The inspector noted that the licensee's actions to date were both appropriate and extensive. The inspector had no further questions at this time.

B. "D" Emergency Diesel Generator (EDG) Start Failure Followup

As discussed in NRC Inspection Reports 354/91-12 (Section 4.3.3.B) and 354/91-14 (Section 4.3.2), the licensee had been pursuing the cause of the May 22, 1991 failure of the "D" EDG to start as required during a surveillance test. In Special Reports 91-03-00 and 91-03-

01 (Supplement), the licensee described the investigation which determined that probable cause of the failure was a lack of fuel boost when the start signal was received. Such a condition could have been caused by either mechanical failure or a mispositioning of the minimum/maximum fuel position switch. The licensee found no mechanical failures and personnel who performed the surveillance tests stated that no repositioning of the fuel boost position switch occurred between test runs. The licensee noted that the EDG had been successfully started eleven successive times since the failure and that the conditions leading to the failure could not be reproduced. Consequently, the licensee concluded that the root cause of the failure was indeterminate. Corrective actions consisted of enhancing the applicable surveillance procedures by including a verification of proper fuel boost position switch position prior to the initiation signal.

The inspector followed closely the licensee's efforts to identify the root cause of the start failure. In addition to activities noted in NRC Inspection Report 354/91-14, the inspector reviewed both special reports and interviewed the EDG system engineer on a frequent basis. The inspector concluded that, although the licensee was unable to definitely identify a root cause, his investigation had been rigorous and expansive, and included the participation of the diesel vendor (Colt Pielstich) and training personnel. The inspector noted that while the investigation did not rule out the possibility of a personnel error, evidence of such was not found. As a preventive measure, the EDG system engineer intends to provide specific training to the equipment operators during their initial or requalification training, as appropriate, to enhance their knowledge of diesel generator performance and controls.

5. EMERGENCY PREPAREDNESS

5.1 Inspection Activity

The inspector reviewed PSE&G's conformance with 10CFR50.47 regarding implementation of the emergency plan and procedures. In addition, licensee event notifications and reporting requirements per 10CFR50.72 and 73 were reviewed.

5.2 Inspection Findings

A. Hurricane Bob Preparations

A hurricane warning was issued for the nearby coastal areas during the period August 18-19, 1991. The site was forecast for high winds and tidal conditions. The licensee made preparations at both Salem and Hope Creek stations for this forecasted condition including:

-- Tracking the hurricane's progress,
-- Reviewing Emergency Classification Guides (ECGs),
-- Implementing abnormal operating procedures,
-- Monitoring meteorological instrumentation,
-- Inspecting the site and all outside areas for non-secure items.

-- Closing water tight doors,
-- Briefing affected personnel on required actions,
-- Verifying operability of offsite and emergency power sources, and
-- Ensuring availability of diesel fuel oil.

The inspectors contacted each control room and discussed preparations with the on-shift senior nuclear shift supervisor. The inspectors also reviewed the associated procedures and ECGs, verified licensee actions, and provided site coverage. The inspectors concluded that the licensee was proactive in their approach to hurricane preparations.

B. Full Participation Onsite Accountability Drill

On Friday, August 23, 1991, the licensee conducted a training drill at the Hope Creek station which included a full scale onsite assembly and accountability scenario, which included all personnel within the protected area at both Hope Creek and Salem. The inspector participated in the drill and observed licensee performance in the Hope Creek Technical Support Center (TSC) from initial staffing to the drill critique. The inspector observed that the drill objectives (demonstrating a coordinated emergency response to simulated plant events and a timely and accurate personnel accountability) were generally met with appropriate coaching and interruptions by the drill controllers. Although some of the emergency response team members were new, team performance appeared good. Personnel demonstrated proactive interest in the drill scenario. For example, although fuel pool cooling pump and heat exchanger status was not reflected on any of the TSC status boards, engineering personnel identified that fuel pool cooling would be lost and provided the emergency duty officer (EDO) with a conservative time estimate for the spent fuel pool temperature to reach the boiling point. A drill critique conducted at the end of the exercise with the command team provided good feedback on the team's performance.

6. SECURITY

6.1 Inspection Activity

PSE&G's conformance with the security program was verified on a periodic basis, including the adequacy of staffing, entry control, alarm stations, and physical boundaries. These inspection activities were conducted in accordance with NRC inspection procedure 71707.

6.2 Inspection Findings

No noteworthy findings were identified.

7. ENGINEERING/TECHNICAL SUPPORT

7.1 Salem

A. Failure and Replacement of Unit 1 Reactor Coolant System Temperature Instrument

On August 11, 1991, the Salem Unit 1 Reactor Coolant System (RCS) 13 loop cold leg narrow range resistance temperature detector (RTD) exhibited signs of intermittent failure, and the channel was declared inoperable. The following day, Salem Instrumentation and Control inspected and tested the RTD and its associated circuitry. The RTD performed satisfactorily and was declared operable. On August 13, 1991, however, the channel again behaved erratically and was again declared inoperable. Subsequently, the circuit was again tested, and the RTD was found to have out-of-specification resistance readings. After the failed RTD was removed, a new spare RTD was resistance checked, documented, and installed while the unit was maintained at power.

A newly installed RTD requires post-installation verification of RTD accuracy to within +/- 0.5 degrees F. At Salem, RTD accuracy is normally verified during plant start-up, at normal operating temperature and zero power, with as near to isothermal equilibrium RCS conditions as possible established prior to the testing. In this case, with the unit operating at power, plant conditions prevented the gathering of the necessary data to utilize the normal methods of RCS RTD accuracy verification. RCS RTD accuracy is significant because the RTD provides an input to two reactor protection system setpoints, over-pressure differential temperature and over-temperature differential temperature. The Westinghouse methodology for protection system setpoints (WCAP 12103) assumes an accuracy of +/- 0.5 degrees F for the temperature inputs.

To confirm the required RTD accuracy after installation, Salem system engineers and PSE&G corporate engineers, with consultation from Westinghouse, developed an alternative analysis to be performed in lieu of the normally performed RTD cross calibration. The method of confirmation consisted of the evaluation and analysis of vendor calibration reports and letters, reactor engineering state point data gathered with the plant at power, data from Westinghouse WCAP 12103, and the pre-installation bench check data of the replacement RTD. The licensee also performed an engineering review and safety evaluation of this method of RTD accuracy verification.

Data was collected at steady state conditions with the new RTD installed and compared to the expected T-cold reading, which was an average T-cold calculated from data from the four previous Unit 1 refueling cycles. The new RTD reading was found to be 0.308 degrees F above the calculated value, which was within one standard deviation of the calculated average and the 0.5 degrees F accuracy requirement. Each of the other three loops also read slightly (from 0.125 to 0.458 degrees F) higher than their calculated value, indicating that some portion of the higher than expected reading of the new RTD was due to a slightly elevated a slightly elevated RCS temperature. The licensee concluded that the process used provided adequate assurance that the new RTD accuracy is within the +/- 0.5 degrees F limit. The 13 loop cold leg narrow range temperature channel was declared operable on August 17, 1991.

The resident inspector followed the licensee resolution of this matter from the time the original RTD was determined to be failed until the analysis of the replacement RTD verified it met the required accuracy standards. Through discussions with the involved PSE&G engineers, attendance at several station management meetings at which the resolution of the problem was planned, and a review of the state point data results and engineering evaluation, the inspector determined that the licensee had used prudent engineering practices and a conservative safety approach in the restoration of the 13 loop T-cold channel. No inadequacies were noted in the licensee's actions or conclusions in this matter.

B. Unit 1 Shutdown Required By Technical Specifications Due to Equipment Failure

On September 5, 1991, an "auto-test fault" alarm was received at 5:09 p.m. from the 1A safeguards equipment cabinet (SEC) at Unit 1. Control room operators attempted to reset the SEC per Operating Procedures; however, the SEC would not reset, rendering it inoperable. Technical Specification (TS) 3.3.2.1 was entered and a unit shutdown from 100% power was commenced per the TS Action requirements. The licensee replaced the installed 1A SEC chassis with a spare chassis and satisfactorily completed a SEC functional test. The unit shutdown was terminated at 50% and the 1A SEC was declared operable at 9:00 p.m. Unit operation was unaffected by this event and the plant was subsequently returned to full power.

The spare chassis was the one which was previously removed from the 1A SEC on August 15, 1991 (See Section 2.2.1.A of this report). The two failed circuit board cards had been replaced and the chassis was functionally tested satisfactorily.

The inspector reviewed the initial conditions prior to the event and the licensee's event response, including immediate actions and conformance with TS requirements. Prior to the 1A SEC failure, multiple safety-related components were made inoperable at 12:42 a.m. on September 5, 1991, due to the tag-out of the No. 12 nuclear service water (SW) header for valve maintenance. Specifically, one of the two intermediate-head safety injection (SI) pumps and one of the two high-head SI pumps were rendered inoperable due to the loss of SW supply flow for the pumps' lubricating oil coolers. The pumps were properly tagged out of service. Additionally, other safety pumps were rendered inoperable (but remained available) due to the loss of SW supply flow to the associated room coolers. The operability of one residual heat removal and two component cooling water system pumps were technically affected by the room coolers being out of service.

The inspector also reviewed impact of the loss of the 1A SEC as related to equipment already made inoperable due to the SW header outage and the TS applicability required actions. The inspector concluded that the appropriate TS Action requirements were properly entered and implemented. The inspector noted that the loss of the 1A SEC rendered the automatic starting and load sequencing for the 1A emergency power source inoperable. However, station abnormal and emergency operating procedures have provisions which direct operators to manually start and load the diesel generators if required under accident conditions. The inspector concluded that redundant system components remained available throughout this event, procedures adequately addressed postulated design basis conditions, and plant safety was not compromised.

7.2 Hope Creek

A. Filtration, Recirculation and Ventilation System (FRVS) Heater Failure Update

During this reporting period the licensee continued the implementation of corrective actions to resolve the issues surrounding the May and July, 1991, FRVS heater failures, as discussed in NRC Inspection Report 354/91-14, Section 7.2.A. The licensee's investigation following the July 6, 1991, fuse failures determined that a build-up of heat inside the panels during the ten hour surveillance run caused a degradation in the current carrying capability of the fuses to a level below the fuse rating. As an interim fix, the heater doors were removed from the recirculation and vent fan panels. Additionally, non-essential heat producing components (e.g., disconnect switch and indicating relays) identified by thermography were de-energized or removed.

On July 30, 1991, the panel vendor, Nutherm International, informed the NRC of a potential deviation in the design safety function of the FRVS panels. Following their evaluation, a 10CFR21 notification was made on August 9, 1991. This notification, however, applied only to the "A" vent and "D" recirculation panels (1AC045 and 1DC043 respectively) and stated that with the doors removed the two panels were operable for both normal and accident conditions.

The inspector noted that test data had been developed by the licensee using two panels as representative of the eight affected panels (1AC043-1FC043, 1AC045 and 1BC045); therefore, the Part 21 determination should have included the other six panels. The licensee agreed and stated that their concerns had been communicated to the vendor by letter dated August 19, 1991. The inspector reviewed licensee and vendor material documenting testing, environmental qualification (EQ), and reportability issues, including Licensee Event Reports (LER) 354/90-07-01 and 91-07-02, and concluded that the licensee's root cause investigation, and corrective actions were thorough and appropriately addressed the outstanding issues concerning 10CFR21 reportability and degraded EQ of certain components. LER 354/91-07-02 appeared to adequately document the resolution of the FRVS heater design versus actual required capacities and the effects of degraded EQ components. While the licensee's initial root cause analysis for the May 1991 fuse failure was inadequate, actions undertaken as a result of the July 1991 failures were extensive, thorough and appropriately managed. The licensee continues pursuing resolution of the Part 21 issue with Nutherm.

B. Reactor Building Ventilation Backdraft Isolation Damper Investigation

In September 1990, the licensee completed an investigation into the spare parts inventory for reactor building supply and exhaust ventilation systems backdraft steam isolation dampers. These are designed to prevent steam from a postulated pipe break from entering non-affected areas of the building through the ventilation ductwork. The results of the licensee's investigation were documented in a letter (SCI-90-0371) dated September 21, 1990, detailing a number of apparent discrepancies relating to environmental qualification (EQ), document inaccuracies, and available spare parts.

An earlier evaluation of the ductwork in February-April, 1988, had determined that none of the 26 pairs of backdraft dampers were included in the EQ program and that substantial further evaluation and documentation would be required to assess the impact on the affected systems. The licensee initiated the appropriate engineering efforts to resolve these issues, efforts which were in progress at the time of the 1990 investigation. The licensee determined that only three of the 26 pairs of backdraft isolation dampers should be (and consequently were entered) in the EQ program. All three were in the Filtration, Recirculation and Ventilation System (FRVS). The equipment qualification maintenance and surveillance (EQMS) information sheet for these dampers, M717-DMPR-004, lacked all the appropriate EQ data. For example, revision 0 of this data sheet specified a ten-year replacement interval, but listed the gasket material as unknown.

The inspector reviewed revision 1 (approved on May 1, 1991) to M717-DMPR-004 and verified that the gasket material was specified and that other discrepancies noted by the 1990 investigation appeared adequately addressed. Justification for the use of four caulk type compounds and a ten-year life time was documented in an April 25, 1991 memorandum to the M717 EQ file. The licensee is currently compiling and will procure the appropriate spare parts for all the backdraft dampers. Completion of procedure enhancements and document updates is tentatively scheduled for the end of 1991. The inspector considered this time frame appropriate considering the minor safety significance of the remaining issues.

8. SAFETY ASSESSMENT/QUALITY VERIFICATION

8.1 Salem

A. Significant Event Response Team (SERT) Report Review

During this report period, the resident inspector staff reviewed two SERT reports that had been prepared for the General Manager-Salem Operations. SERT Report No. SSR 91-03 reported the investigation of the Salem Unit 1 1B Vital Bus Undervoltage (UV) Relay events of June 6 and June 13,1991,(see NRC Inspection Report 272/91-15, Section 4.3.2.C) and SERT Report No. SSR 91-04 assessed the Salem Unit 1 Reactor Trip and Lightning Strike of June 16,1991 (see NRC Inspection Report 272 & 311/91-19, Section 2.2.3).

The reports documented the findings, conclusions and recommendations of the two SERTs that had been formed in accordance with PSE&G Nuclear Administrative Procedure NC.NA-AP.ZZ-0061(Q),"Significant Event Response Team Management." This procedure states that the purpose of a SERT is to provide for "independent assessment of selected events, trends or certain repetitive situations" and to "ensure that all relevant aspects of an event or situation have been considered and appropriate corrective actions identified to prevent recurrence." The inspectors' review of the two reports revealed that both SERTs had accomplished these functions and adequately reviewed each event; a good questioning attitude and a proper safety perspective was noted.

The inspectors identified one weakness in SERT Report No. SSR 91-03. The SERT did not identify the fact that a certain human engineering deficiency (the location of the relays and the lack of test jacks for surveillance testing of the UV relays), had a direct impact on the June 6, 1991 event, and was previously identified by Salem technicians. The correction of this deficiency might have prevented these events. When informed by the inspector of this finding, the licensee acknowledged that the information should have been considered in the report and that it would be considered in the resolution of the UV relay testing concern.

Other than the one identified weakness, the inspector concluded that the SERT process had been effectively utilized in the assessment of the two events.

8.2 Hope Creek

A. Hope Creek Comprehensive Scram Review

Following the May 7, 1991 unplanned scram at Hope Creek (NRC Inspection 354/90-12), a team from Onsite Safety Review, Offsite Safety Review, Station Quality Assurance and Human Performance Enhancement System reviewed the 12 scram events since August 26, 1988. The licensee's team used the Management Oversight and Risk Tree (MORT) process. This team assessed event-specific and management-related factors that contributed to or allowed the scrams to happen.

The team's conclusions and recommendations focused on the following:

-- Establishing scram reduction responsibility,

-- Communicating scram-specific quality expectations,

-- Establishing employee involvement and feedback processes,

-- Re-emphasizing the Scram and Power Reduction Elimination Committee (SPRE),

-- Ensuring that training and job content are in a climate of procedure adherence and routinized work,

- -- Enhancing balance-of-plant maintenance, and effecting latent error reduction in all maintenance,
- -- Implementing Significant Event Review Team (SERT) recommendations,
- -- Balancing plant versus people-oriented corrective actions, and
- -- Reviewing the value of the action tracking system.

The inspector reviewed the final report dated July 29, 1991. The inspector concluded that the licensee's assessment, conclusions and recommendations appeared appropriate. Overall, this effort was thorough, well managed, and effectively conducted.

9. LICENSEE EVENT REPORTS (LER), PERIODIC AND SPECIAL REPORTS FOLLOWUP

PSE&G submitted the following licensee event reports, and special and periodic reports, which were reviewed for accuracy and evaluation adequacy.

Special and Periodic Reports

- -- Semi-Annual Fitness For Duty Performance Data dated August 19, 1991.
- -- Salem and Hope Creek Monthly Operating Reports for July 1991.
- -- Salem Unit 1 Inservice Inspection Activities for the Ninth Refueling Outage (NLR-N91128).
- -- Salem and Hope Creek Semi-Annual Effluent Release Reports for period January 1 to June 30, 1991.
- -- Salem Unit 1 Special Report 91-2 addressed the "A" Reactor Trip Breaker failure of July 25, 1991 (See NRC Inspection Report 272/91-19, Section 2.2.1.B).
- -- Hope Creek Special Report 91-03-01 (Supplement) See Section 4.3.2.B

No unacceptable conditions were noted.

Salem LERs

Unit 1

LER 91-07, Revision 1 updated an event regarding automatic starting of the motor driven auxiliary feedwater pumps during an outage. The event was reviewed in NRC Inspection 272/91-05. No inadequacies were noted relative to this LER.

LER 91-08, Revision 1 updated a reactor protection system actuation while shut down in Mode 5. The event was reviewed in NRC Inspection 272/91-05. No inadequacies were noted relative to this LER.

LER 91-21, Supplement 1, addressed an additional containment penetration overcurrent protection device Technical Specification 3/4.8.3.1 noncompliance discovered after the submission of the original LER (See NRC Inspection Report 272/91-19, Section 4.3.1.C). No inadequacies were noted relative to this supplement.

LER 91-24 addressed the Unit 1 reactor trip due to a lightning strike on June 16, 1991, which was discussed in NRC Inspection Report 272/91-19, Section 2.2.3. No inadequacies were noted relative to this LER.

LER 91-25 concerned a radiation monitor spike (1R45C) and containment ventilation isolation on July 27, 1991, due to equipment failure. No inadequacies were noted relative to this LER.

LER 91-26 addressed the No. 13 steam generator pressure protection channel inoperability resulting from the lead-lag switch incorrect setting, which is discussed in Section 4.3.1.A of this report.

LER 91-27 (See Section 2.2.1.A)

Unit 2

LER 91-06, Revision 1 updated an event regarding failure of a control room radiation monitor due to equipment failure. The event was reviewed in NRC Inspections 311/91-05, 09. No inadequacies were noted relative to this LER.

LER 91-08 addressed the vital 4KV bus undervoltage relays that were found to have setpoints below the Technical Specification minimum allowed value (see NRC Inspection Report 311/91-15, Section 4.3.2.E). No inadequacies were noted relative to the LER.

LER 91-09 addressed the spurious start of the No. 21 motor driven auxiliary feedwater pump of June 30, 1991, which was reviewed in NRC Inspection Report 311/91-19, Section 2.2.1.A. No inadequacies were noted relative to the LER.

LER 91-11 addressed the July 30, 1991, discovery that the non-radioactive liquid waste discharge radiation monitor system (RMS) channel 2R37 setpoint was not in compliance with Technical Specification 3.3.3.8. Higher capacity pumps had been installed in the system in June 1985, and the corresponding RMS setpoint changes were not implemented until the July 30, 1991 discovery. The licensee attributed the root cause of the event to inadequate design review and subsequently completed a design change to modify the 2R37 setpoint. The inspector reviewed the licensee actions, determined them to be adequate, and noted no inadequacies relative to the LER.

LER 91-10 concerned a radiation monitor spike (2R45C) and containment ventilation isolation on July 23, 1991, due to equipment failure. No inadequacies were noted relative to this LER.

Hope Creek LERs

LER 91-07-01 (See Section 7.2.A).

LER 91-07-02 (See Section 7.2.A).

LER 91-16 described an isolation of the Reactor Core Isolation Cooling (RCIC) System due to a spurious signal from the steam leak detection system. A NUMAC temperature sensor card associated with the inboard RCIC steam isolation valve failed high; the resulting signal caused the valve to fully close. After determining that the signal was spurious, repairs were initiated and RCIC returned to operable status. The safety significance of this first-time event was minimal. No significant discrepancies were noted in this LER.

10. EXIT INTERVIEWS/MEETINGS

10.1 Resident Exit Meeting

The inspectors met with Mr. C. Vondra and Mr. R. Hovey and other PSE&G personnel periodically and at the end of the inspection report period to summarize the scope and findings of their inspection activities.

Based on Region I review and discussions with PSE&G, it was determined that this report does not contain information subject to 10 CFR 2 restrictions.

10.2 Specialist Entrance and Exit Meetings

Date(s)	Subject	Inspection Report No.	Reporting Inspector
8/12-20/91	Emergency Preparedness	272&311/91-24; 354/91-17	Amato
8/19-23/91	Radiological Control	354/91-15	Mann
8/26-30/91	Environmental Monitoring	272&311/91-22; 354/91-18	Peluso

10.3 Hope Creek Meeting

A. Motor Operated Valve (MOV) Program Enforcement Conference

An enforcement conference was held on September 9, 1991, in the NRC Region I King of Prussia office to discuss a number of issues arising from NRC Inspection 354/91-80, conducted July 15-19, 1991, at Hope Creek, pertaining to the operator, testing, and safety evaluation of Motor Operated Valves as described in NRC Generic Letter 89-10. Attachment 1 is a list of attendees, and Attachments 2 and 3 describe the licensee's presentation. The NRC's conclusions from this conference will be provided to the licensee in a separate correspondence.

ATTACHMENT 1

ENFORCEMENT CONFERENCE
LIST OF ATTENDEES
September 9, 1991

NUCLEAR REGULATORY COMMISSION

J. Wiggins, Deputy Director, Division of Reactor Projects (DRP)
W. Lanning, Deputy Director, Division of Reactor Safety (DRS)
R. Blough, Chief, Reactor Projects Branch 2, DRP
J. White, Chief, Reactor Projects Section No. 2A, DRP
P. Eapen, Chief, Systems Section, DRS
J. Durr, Chief, Engineering Branch, DRS
K. Lathrop, Resident Inspector
R. Matakas, Investigator, Office of Investigations
D. Holody, Enforcement Officer
J. Yerokun, Project Engineer
B. Westreich, Reactor Engineer
K. Smith, Regional Counsel
W. Butler, Project Director, NRR
S. Dembek, Project Manager, NRR
J. Colaccino, Mechanical Engineer, NRR
E. Sullivan, Section Chief, NRR
T. Scarbrough, Senior Mechanical Engineer, NRR

PUBLIC SERVICE ELECTRIC AND GAS COMPANY

S. Miltenberger, Vice President & Chief Nuclear Officer
T. Crimmins, Jr., Vice President - Nuclear Engineering
J. Hagan, General Manager - Hope Creek
L. Reiter, General Manager - QA & Nuclear Safety Review
D. Jagt, Manager - Nuclear Engineering Design
G. Englert, Jr., Nuclear Engineering Standards Manager
J. Ranalli, Mechanical Engineering Manager
R. Brown, Principal Engineer - Licensing
F. Thomson, Manager - Nuclear Licensing & Regulation
K. Suomi, Senior Nuclear Maintenance Supervisor
R. Binz, Principal Engineer

OTHER

M. Sesok, Atlantic Electric Site Representative
C. Dell, Nuclear Engineer, State of New Jersey

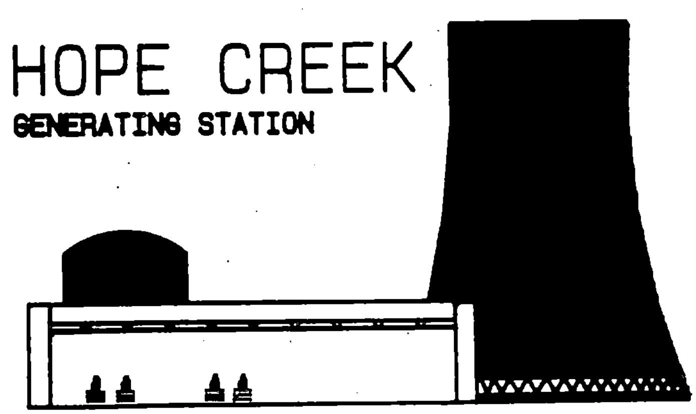
Public Service Electric and Gas Company

ATTACHMENT 2

NRC ENFORCEMENT CONFERENCE

MOV PROGRAM

SEPTEMBER 9, 1991



AGENDA

MOV PROGRAM

INTRODUCTION/OVERVIEW	T. M. CRIMMINS
SAFETY ASSESSMENT RESULTS	J. A. RANALLI
PROGRAM DESCRIPTION	J. A. RANALLI
PROGRAM DESCRIPTION DEVIATION	J. A. RANALLI
SUPPLEMENT 3 RESPONSE	J. A. RANALLI
TORQUE SWITCH SETTING	J. J. HAGAN
VENDOR INFORMATION CONTROL	G. E. ENGLERT
PSE&G ASSESSMENT OF POTENTIAL VIOLATIONS	F. X. THOMSON
SUMMARY	T. M. CRIMMINS

MOV PROGRAM

INTRODUCTION/OVERVIEW

OVERVIEW OF INSPECTION FINDINGS

- POTENTIAL DEVIATION
  - THE LICENSEE HAD NOT ESTABLISHED A DETAILED GL 89-10 PROGRAM DESCRIPTION BY JANUARY 1, 1991 AS COMMITTED

- POTENTIAL VIOLATIONS
  - PROVIDING INACCURATE AND INCOMPLETE INFORMATION TO THE NRC IN LICENSEE LETTER DATED MARCH 8, 1991 CONTRARY TO 10CFR50.9
  - MODIFICATION OF MOV TORQUE SWITCH SETTINGS WITHOUT AN ENGINEERING OR SAFETY EVALUATION
  - FAILURE TO REVIEW, EVALUATE AND INCORPORATE CERTAIN VENDOR INFORMATION WHICH PROVIDED INFORMATION FOR THE MAINTENANCE OF SAFETY-RELATED MOVs

MOV PROGRAM

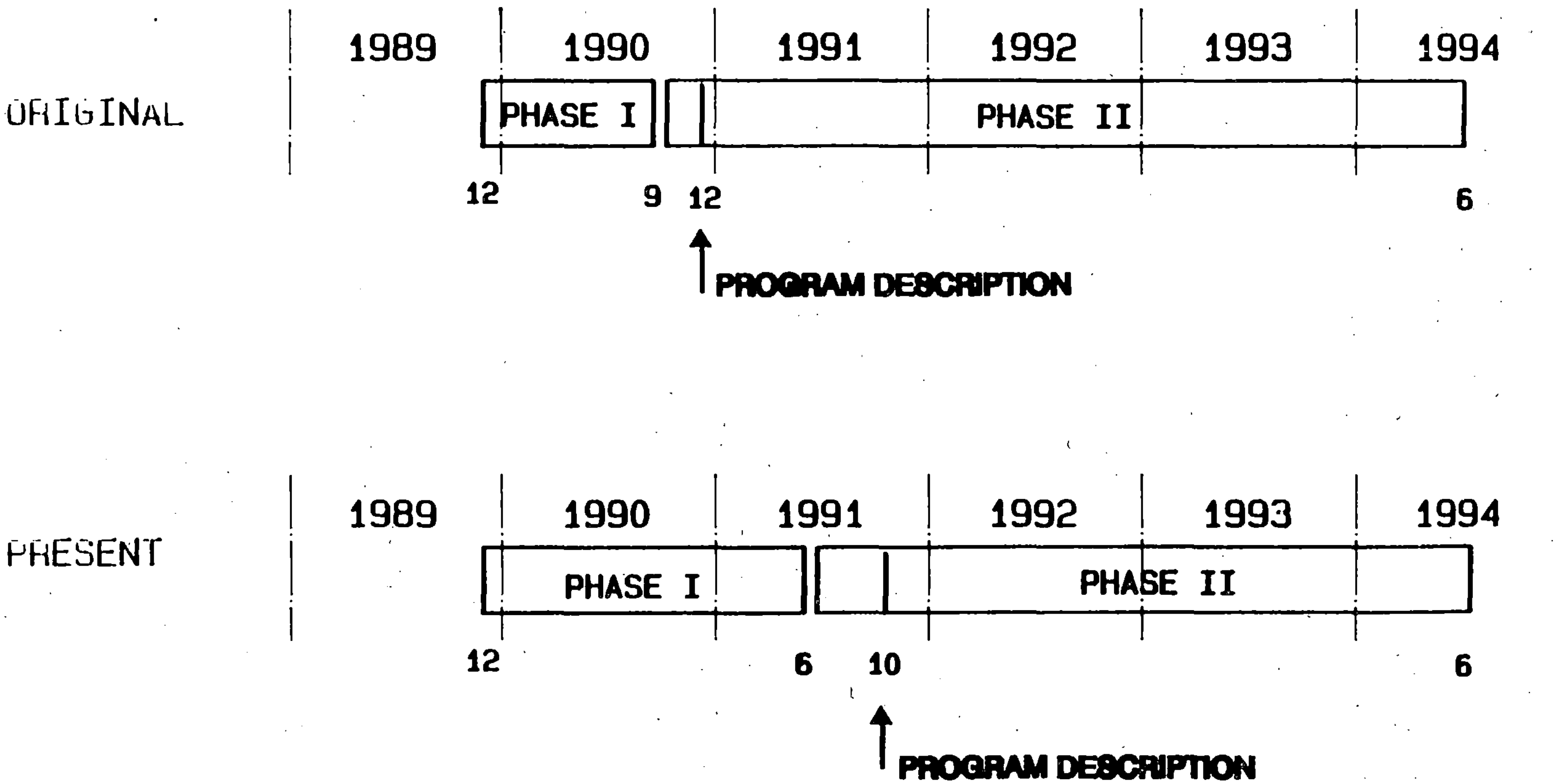
OVERALL SAFETY ASSESSMENT

- DETAILED REVIEW OF SUPPLEMENT 3 ANALYSIS VERIFIES VALVES WILL CLOSE UNDER DBA CONDITIONS

- SUBSTANTIAL MARGINS EXIST DUE TO TORQUE SWITCH BYPASS CIRCUITRY
  - ENHANCES VALVE THRUST CAPABILITY TO ACHIEVE DISK-TO-SEAT OVERLAP
  - ALLOWS ACCOMODATION OF HIGHER VALVE FACTORS
  - REMOVES RATE OF LOADING AS A CONCERN FOR THE DURATION OF BYPASS
  - VALVE ASSEMBLY HAS ADEQUATE STRUCTURAL CAPABILITY TO WITHSTAND STRESSES

- SUBSEQUENT EVALUATIONS CONFIRM OPERABILITY CONCLUSIONS

MOV PROGRAM SCHEDULE



91F61-21

MOV PROGRAM

PROGRAM DESCRIPTION

PHASE I ACTIVITIES

- DEVELOP ENGINEERING REQUIREMENTS AND IMPLEMENTATION PLANS

- IDENTIFY 15 MOVs FOR DETAILED ANALYSIS
  - 9 SALEM VALVES
  - 6 HOPE CREEK VALVES

- ASSESS EXISTING MOV PROGRAM AGAINST GL 89-10 RECOMMENDATIONS

- IDENTIFY SCOPE ITEMS FOR INCLUSION IN PHASE II PROGRAM

MOV PROGRAM

PROGRAM DESCRIPTION

LESSONS LEARNED - PHASE I

- PRINCIPAL ENHANCEMENTS IDENTIFIED FOR PHASE II INCLUDE:
  - ENHANCEMENT OF CURRENT NUCLEAR DEPARTMENT POLICY GOVERNING OVERALL MAINTENANCE PROGRAM
  - DEVELOPMENT OF PROGRAMMATIC STANDARDS FOR CONTROL OF FUTURE MOV ACTIVITIES
  - CONSOLIDATION AND RECONCILIATION OF EXISTING DATA SOURCES
  - DEVELOPMENT OF IMPROVED METHODS AND PROCEDURES FOR MOV TESTING AND MAINTENANCE

MOV PROGRAM

PROGRAM DESCRIPTION

PHASE II ACTIVITIES

- IMPLEMENT POLICY AND FORMAL PROGRAM REQUIREMENTS TO MEET GL 89-10 RECOMMENDATIONS
  - DEVELOP PROGRAM IMPLEMENTING PROCEDURES
    - ▲ MANAGEMENT POLICY
    - ▲ DATA COLLECTION/RECONCILIATION
    - ▲ FUNCTIONAL EVALUATIONS
    - ▲ CORRECTIVE ACTION
    - ▲ TRENDING
  - DEVELOP IMPROVED MAINTENANCE METHODS
    - ▲ TESTING AND MAINTENANCE

- ESTABLISH DETAILED SCHEDULES

- COMPLETION BY JUNE 1994

MOV PROGRAM

PROGRAM DESCRIPTION

MOV PROGRAM PROJECT PLAN

- PROJECT CONTROLS AND FUNCTIONAL RESPONSIBILITIES
- ORGANIZATIONAL MATRIX

MOV PROGRAMMATIC STANDARD NC.DE-PS.ZZ-0033 (Q)

- APPENDICES OF PROGRAMMATIC STANDARD
  - APPENDIX A1 VALVE POPULATION
  - APPENDIX A2 VALVE PRIORITIZATION
  - APPENDIX A3 WALKDOWN DATA COLLECTION
  - APPENDIX A4 OPERATING CONDITION EVALUATION
  - APPENDIX A5 ELECTRICAL CAPABILITY REVIEW
  - APPENDIX A6 MECHANICAL CAPABILITY REVIEW
  - APPENDIX A7 DIAGNOSTIC TEST DATA REVIEW
  - APPENDIX A8 CAPABILITY ASSESSMENT
  - APPENDIX A9 DATABASE SYSTEM
  - APPENDIX A10 MOVATS DIAGNOSTIC DATA CONSOLIDATION
  - APPENDIX A11 SOFTWARE REQUIREMENTS SPECIFICATIONS
  - APPENDIX A12 DATA COLLECTION SPECIFICATION
- APPROVAL OF THESE DOCUMENTS IN TOTAL WILL BE ACCOMPLISHED BY 10/31/91
- ONGOING ACTIVITIES:
  - DATA GATHERING
  - MOV EVALUATIONS TO SUPPORT MAINTENANCE ACTIVITIES

MOV PROGRAM

PROGRAM DESCRIPTION DEVIATION

NRC FINDING

- PSE&G FAILED TO MEET ITS COMMITMENT TO ESTABLISH AN APPROVED PROGRAM DESCRIPTION BY JANUARY 1, 1991

PSE&G RESPONSE

- WE AGREE WITH THE DEVIATION AS STATED

ROOT CAUSE OF DEVIATION

- WE FAILED TO RECOGNIZE AND ADDRESS THE OVERCOMMITMENTS ON OUR VALVE ENGINEERING RESOURCES IN A TIMELY MANNER
  - TURNOVER OF KEY PERSONNEL
  - DIVERSION OF RESOURCES TO OTHER SIGNIFICANT ISSUES
- WE FAILED TO UTILIZE OUR MONITORING AND CONTROL PROCESSES FOR TRACKING OUR GL 89-10 PROGRAM DESCRIPTION COMMITMENT

10 MONTH SLIP IN SCHEDULE RESULTED

MOV PROGRAM

PROGRAM DESCRIPTION DEVIATION

CORRECTIVE ACTIONS

- VICE PRESIDENT - NUCLEAR ENGINEERING (VPNE) HAS COUNSELED ALL MANAGEMENT PERSONNEL DIRECTLY INVOLVED
- VPNE LETTER ISSUED TO ALL NUCLEAR ENGINEERING PERSONNEL REITERATING EXPECTATIONS
  - CONTROL OF COMMITMENTS
  - SCHEDULE ADHERENCE
  - TIMELY RECONCILIATION OF RESOURCE ISSUES
  - REQUESTED MANAGERS TO REVIEW OTHER REGULATORY PROGRAMS TO ENSURE SIMILAR PROBLEMS DO NOT EXIST
- AN INDEPENDENT REVIEW HAS BEEN INITIATED TO ASSESS THE EFFECTIVENESS OF OUR REGULATORY COMMITMENT TRACKING PROCESS (12/31/91)
- PROJECT PLAN WAS RE-ESTABLISHED IN OCTOBER 1990
  - PHASE I ACTIVITIES COMPLETE
  - PHASE II ACTIVITIES UNDERWAY
  - FINAL COMPLETION IN JUNE 1994

MOV PROGRAM

SUPPLEMENT 3 RESPONSE

NRC FINDINGS

PSE&G STATED THAT RESPONSE TO SUPPLEMENT 3 WAS BASED ON COMPREHENSIVE EVALUATION OF THRUST REQUIREMENTS. THE FOLLOWING DISCREPANCIES AND INACCURACIES WERE NOTED BY THE NRC DURING THE INSPECTION:

- THRUST VALUES REPORTED WERE NOT DERIVED FROM DETAILED EVALUATIONS BUT WERE OBTAINED FROM VENDOR DATA SHEETS

- PSE&G HAD 3 SETS OF EVALUATIONS INDICATING THAT REPORTED THRUST VALUES MIGHT BE NON-CONSERVATIVE

- LACK OF TECHNICAL JUSTIFICATION FOR USE OF NON-CONSERVATIVE DISK AND STEM FRICTION FACTORS

- REPORTED THRUST MARGINS DID NOT INCLUDE CONSIDERATION OF INSTRUMENT UNCERTAINTY AND RATE OF LOADING

- REPORTED THRUST VALUES WERE NOT BASED ON DESIGN BASIS DP TESTING. REPORTED THRUST VALUES WERE BASED ON TORQUE SWITCH TRIP, WHEREAS TORQUE SWITCH IS BYPASSED DURING ACCIDENT CONDITIONS

MOV PROGRAM

SUPPLEMENT 3 RESPONSE

PSE&G RESPONSE

- RESPONSE WAS "COMPLETE AND ACCURATE" BASED UPON OUR ENGINEERING JUDGEMENT AND EVALUATION

- DIFFERING PROFESSIONAL OPINIONS EXIST WITHIN THE INDUSTRY ON CERTAIN TECHNICAL ISSUES

- WE FAILED TO ADEQUATELY COMMUNICATE THE BASIS FOR OUR CONCLUSION OF OPERABILITY
  - SUBSEQUENT RE-EVALUATION OF DATA CONFIRMED OPERABILITY CONCLUSION STATED IN OUR RESPONSE

- VIOLATION OF 10CFR50.9 NOT JUSTIFIED

MOV PROGRAM

SUPPLEMENT 3 RESPONSE

NRC FINDINGS

- THRUST VALUES REPORTED WERE NOT DERIVED FROM DETAILED EVALUATIONS BUT WERE OBTAINED FROM VENDOR DATA SHEETS

- PSE&G HAD 3 SETS OF DATA INDICATING THAT REPORTED THRUST VALUES MIGHT BE NON-CONSERVATIVE

MOV PROGRAM

SUPPLEMENT 3 RESPONSE

PSE&G RESPONSE

REPORTED THRUST VALUES WERE TAKEN FROM VENDOR VALVE DATA SHEETS

- REPRESENTED "BEST AVAILABLE" DESIGN INFORMATION

SELECTION OF VENDOR THRUST VALUES FOR USE IN SUPPLEMENT 3 WAS BASED ON ENGINEERING REVIEW OF AVAILABLE INFORMATION

- VENDOR DATA
- PHASE I RESULTS
- SAFETY ASSESSMENT

ENGINEERING REVIEW OF AVAILABLE INFORMATION IDENTIFIED THE FOLLOWING:

- VENDOR THRUST VALUES WERE CONSERVATIVELY DERIVED
  - DIFFERENTIAL PRESSURE
  - UNDERVOLTAGE
- PHASE I RESULTS INCLUDED CONSERVATIVE ASSUMPTIONS
  - WELD END PREP vs SEAT DIAMETER
  - BYPASS CIRCUITRY NOT INCLUDED
- SAFETY ASSESSMENT CREDITED BYPASS CIRCUITRY FOR PROVIDING SIGNIFICANT THRUST CAPABILITY BEYOND MINIMUM REQUIRED THRUST

MOV PROGRAM

SUPPLEMENT 3 RESPONSE

PSE&G RESPONSE (CONT)

DRAFT ENGINEERING RESPONSE TO SUPPLEMENT 3 ESTABLISHED THE BASIS FOR OUR OPERABILITY CONCLUSION

- RECONCILED DIFFERENCES IN MINIMUM THRUST VALUES ON BASIS OF CONSERVATIVE ANALYSIS ASSUMPTIONS
- CONCLUDED DESIGN BASIS VALUES (i.e. VENDOR DATA) REPRESENTED "BEST AVAILABLE" INFORMATION
- IDENTIFIED TORQUE SWITCH BYPASS CIRCUITRY AS PROVIDING ADDITIONAL ASSURANCE OF VALVE CLOSURE CAPABILITY UNDER DBA CONDITIONS
  - ALLOWS THRUST UP TO MOTOR RATED TORQUE
  - GREATLY EXCEEDS TORQUE SWITCH TRIP THRUST

WE FAILED TO ADEQUATELY COMMUNICATE THIS BASIS IN OUR RESPONSE TO SUPPLEMENT 3

BASED ON RE-EVALUATION OF DATA AVAILABLE PRIOR TO 3/8/91 AND THE OPERABILITY BASIS DESCRIBED IN OUR DRAFT RESPONSE, WE CONCLUDE THAT OUR OPERABILITY DETERMINATION PROVIDED IN RESPONSE TO SUPPLEMENT 3 REMAINS VALID

- RECENT EVALUATIONS CONTINUE TO CONFIRM THE OPERABILITY OF THESE VALVES

MOV PROGRAM

SUPPLEMENT 3 RESPONSE

NRC FINDING

LACK OF TECHNICAL JUSTIFICATION FOR USE OF NON-CONSERVATIVE DISK AND STEM FRICTION FACTORS

PSE&G RESPONSE

- CURRENT DISK AND STEM FRICTION FACTORS ARE THE RESULT OF BEST AVAILABLE VENDOR RECOMMENDATIONS
- OUR PHASE II PROGRAM WILL RE-EVALUATE THE VALVE FACTORS TO BE USED
- DIFFERING PROFESSIONAL OPINIONS EXIST WITHIN THE INDUSTRY ON THE EXACT VALUES TO BE USED
- 15% $\pm$ 5% MARGIN IS ASSIGNED TO ACCOUNT FOR ENGINEERING UNCERTAINTIES AND EQUIPMENT INACCURACIES
- BYPASS OF TORQUE SWITCH ENHANCES VALVE THRUST CAPABILITY, THEREBY ALLOWING HIGHER VALVE FACTORS TO BE ACCOMODATED

MOV PROGRAM

SUPPLEMENT 3 RESPONSE

NRC FINDING

REPORTED THRUST MARGINS DID NOT INCLUDE CONSIDERATION OF INSTRUMENT UNCERTAINTY AND RATE OF LOADING

PSE&G RESPONSE

- REPORTED THRUST MARGINS WERE NOT ADJUSTED FOR INSTRUMENT INACCURACIES NOR DID THEY ADDRESS RATE OF LOADING EFFECTS
- 15% ± 5% MARGIN IS ASSIGNED TO ACCOUNT FOR INSTRUMENT INACCURACIES AND ENGINEERING UNCERTAINTIES
- ACTUAL MARGINS ARE GREATER CONSIDERING TORQUE SWITCH BYPASS CIRCUITRY
  - ACTUAL DEVELOPED THRUST COULD BE EQUIVALENT TO MOTOR RATED TORQUE
  - DEFEAT OF BYPASS OCCURS AFTER DISK-TO-SEAT OVERLAP IS ACCOMPLISHED
- RATE OF LOADING IS NOT A CONCERN WHEN THE TORQUE SWITCH IS BYPASSED
- FINAL DISPOSITION OF THESE ISSUES IS REQUIRED UNDER PSE&G PHASE II PROGRAM

MOV PROGRAM

SUPPLEMENT 3 RESPONSE

NRC FINDING

REPORTED THRUST VALUES WERE NOT BASED ON DESIGN BASIS DP TESTING. REPORTED THRUST VALUES WERE BASED ON TORQUE SWITCH TRIP, WHEREAS TORQUE SWITCH IS BYPASSED DURING ACCIDENT CONDITIONS.

PSE&G RESPONSE

- REPORTED THRUST VALUES REFLECTED MOST RECENT DIAGNOSTIC RESULTS OBTAINED UNDER STATIC CONDITIONS
  - REPORTED THRUST VALUES WERE BASED ON TORQUE SWITCH TRIP
- IN-SITU DESIGN BASIS DP TESTING IS NOT PRACTICABLE FOR THESE VALVES
- BYPASS CIRCUITRY PROVIDES ADDITIONAL CAPABILITY TO CLOSE VALVES
  - ALLOWS THRUST UP TO MOTOR RATED TORQUE CAPABILITY
  - MOTOR RATED TORQUE IS MUCH GREATER THAN TORQUE SWITCH TRIP THRUST
  - DIAGNOSTIC TRACES SHOW SUBSTANTIAL DISK-TO-SEAT OVERLAP WHEN BYPASS IS DEFEATED

MOV PROGRAM

SUPPLEMENT 3 RESPONSE

ACTIONS TAKEN TO DATE

- RE-EVALUATED EXISTING DATA TO CONFIRM NO OPERABILITY ISSUE

- AN INDEPENDENT ASSESSMENT HAS BEEN CONDUCTED BY OUR NUCLEAR SAFETY REVIEW GROUP TO EVALUATE THE CIRCUMSTANCES LEADING TO OUR SUPPLEMENT 3 RESPONSE

- VPNE HAS DISCUSSED THE ISSUE WITH DIRECT REPORTS AND REITERATED HIS EXPECTATIONS RELATIVE TO COMMUNICATIONS WITH THE NRC

ACTIONS TO BE COMPLETED

- GL 89-10 SUPPLEMENT 3 RESPONSE WILL BE RESUBMITTED BY 9/30/91 TO CLARIFY THE BASIS FOR OUR OPERABILITY DETERMINATION

- LETTER TO BE ISSUED TO ALL NUCLEAR DEPARTMENT PERSONNEL
  - TO EMPHASIZE OBLIGATION TO PROVIDE CLEAR INFORMATION IN COMMUNICATIONS WITH NRC
  - TO EMPHASIZE THAT TECHNICAL ISSUES MUST BE ADEQUATELY EXPLAINED
  - LETTER WILL ALSO BE DISCUSSED WITH PERSONNEL THROUGHOUT THE NUCLEAR DEPARTMENT VIA "ROLLDOWN" BY MANAGEMENT/SUPERVISION

- TRAINING TO BE HELD WITH ALL LICENSING DEPARTMENT PERSONNEL TO ENSURE EXPECTATIONS/RESPONSIBILITIES ARE UNDERSTOOD RELATIVE TO COMMUNICATIONS WITH THE NRC

MOV PROGRAM

SUPPLEMENT 3 RESPONSE

SUMMARY

- OUR RESPONSE TO GENERIC LETTER 89-10, SUPPLEMENT 3 WAS COMPLETE AND ACCURATE BASED ON ENGINEERING JUDGEMENT AND EVALUATION AND THEREFORE NOT IN VIOLATION OF 10CFR50.9
- WE FAILED TO ADEQUATELY COMMUNICATE THE BASIS FOR OUR OPERABILITY CONCLUSION
  - NO INTENT TO MISLEAD THE NRC
- SUBSTANTIAL MARGINS EXIST TO ASSURE PROPER VALVE FUNCTION UNDER DESIGN BASIS ACCIDENT CONDITIONS
- TECHNICAL CONCERNS RAISED IN GENERIC LETTER 89-10 WILL BE ADDRESSED UNDER OUR PHASE II PROGRAM
  - DIAGNOSTIC EQUIPMENT INACCURACIES
  - RATE OF LOADING
  - APPROPRIATE VALVE FACTORS
- ACTIVELY INVOLVED IN INDUSTRY GROUPS FOLLOWING MOV ISSUES
- WE WILL RESUBMIT OUR SUPPLEMENT 3 RESPONSE TO CLARIFY THE BASIS FOR OUR OPERABILITY DETERMINATION BY 9/30/91

MOV PROGRAM

TORQUE SWITCH SETTING

NRC FINDING

- MODIFICATION OF MOV TORQUE SWITCH SETTINGS WITHOUT A DOCUMENTED ENGINEERING OR SAFETY EVALUATION

DESCRIPTION OF DEFICIENCY

- LIMITER PLATES REMOVED FROM 2 RCIC VALVE MOTOR OPERATORS
- TORQUE SWITCH SETTINGS INCREASED TO OBTAIN REQUIRED THRUST
- NO DOCUMENTED ENGINEERING EVALUATION PERFORMED

PSE&G RESPONSE

- WE AGREE WITH THE VIOLATION AS STATED

MOV PROGRAM

TORQUE SWITCH SETTING

EVALUATION OF DEFICIENCY

- SWITCH SETTINGS UNDER ADMINISTRATIVE CONTROL OF MAINTENANCE PROCEDURE
  - SYSTEM ENGINEER APPROVAL WITHIN PROCEDURE
  - LIMITORQUE AND VALVE MANUFACTURER CONCURRENCE OBTAINED
  - SWITCH SETTING INTENDED TO PROVIDE DESIGN BASIS SEATING THRUST
  - LIMITER PLATE SETTING BASED ON VENDOR CALCULATION
    - ▲ LESS ACCURATE THAN DIRECT MEASUREMENT

- NO DOCUMENTATION OF EVALUATIONS PERFORMED

MOV PROGRAM

TORQUE SWITCH SETTING

ROOT CAUSE

- PAST PROCEDURAL CONTROLS FAILED TO ADDRESS DOCUMENTATION REQUIREMENTS

CORRECTIVE ACTIONS

- MAINTENANCE PROCEDURES REVISED TO REQUIRE DEFICIENCY REPORT (DR) INITIATION TO RE-EMPHASIZE DOCUMENTATION REQUIREMENT
  - DR REQUIRES 50.59 PROCESS

- COMPLETED REVIEW OF POST-STARTUP RECORDS FOR SIMILAR OCCURRENCES
  - 13 AFFECTED VALVES IDENTIFIED
  - ENGINEERING DISCREPANCY EVALUATION PROCESS USED TO EVALUATE
    - ▲ INITIAL OPERABILITY SCREENING
    - ▲ PRIORITIZED BASED ON SAFETY SIGNIFICANCE (PRA)
    - ▲ ENGINEERING EVALUATION BASED ON LIMITORQUE MAINTENANCE UPDATE 89-01
  - INITIAL SCREENING AND EVALUATION INDICATES NO SAFETY SIGNIFICANCE FOR LIMITER PLATE REMOVAL

- COMPLETION OF PRE-STARTUP DOCUMENTATION REVIEWS BY NOVEMBER 15, 1991

MOV PROGRAM

TORQUE SWITCH SETTING

PRESENT PROGRAM FOR CONTROL OF TORQUE SWITCH SETTINGS

- DEFICIENCY REPORT REQUIRED IN ORDER TO EXCEED SPECIFIED SETTINGS
  - REQUIRES DOCUMENTED ENGINEERING EVALUATION
  - ENSURES 50.59 IS ADDRESSED

MOV PROGRAM

TORQUE SWITCH SETTING

SUMMARY

- OUR PAST EVALUATIONS WERE NOT FORMALLY DOCUMENTED

- NO SAFETY SIGNIFICANCE

- RECORDS ARE BEING REVIEWED FOR SIMILAR OCCURRENCES AT BOTH STATIONS
  - ENGINEER DISCREPANCY EVALUATION PROCESS WILL BE INITIATED FOR SIMILAR OCCURRENCES IDENTIFIED

- REVISED PROCEDURES REQUIRE DOCUMENTATION OF FUTURE ADJUSTMENTS

MOV PROGRAM

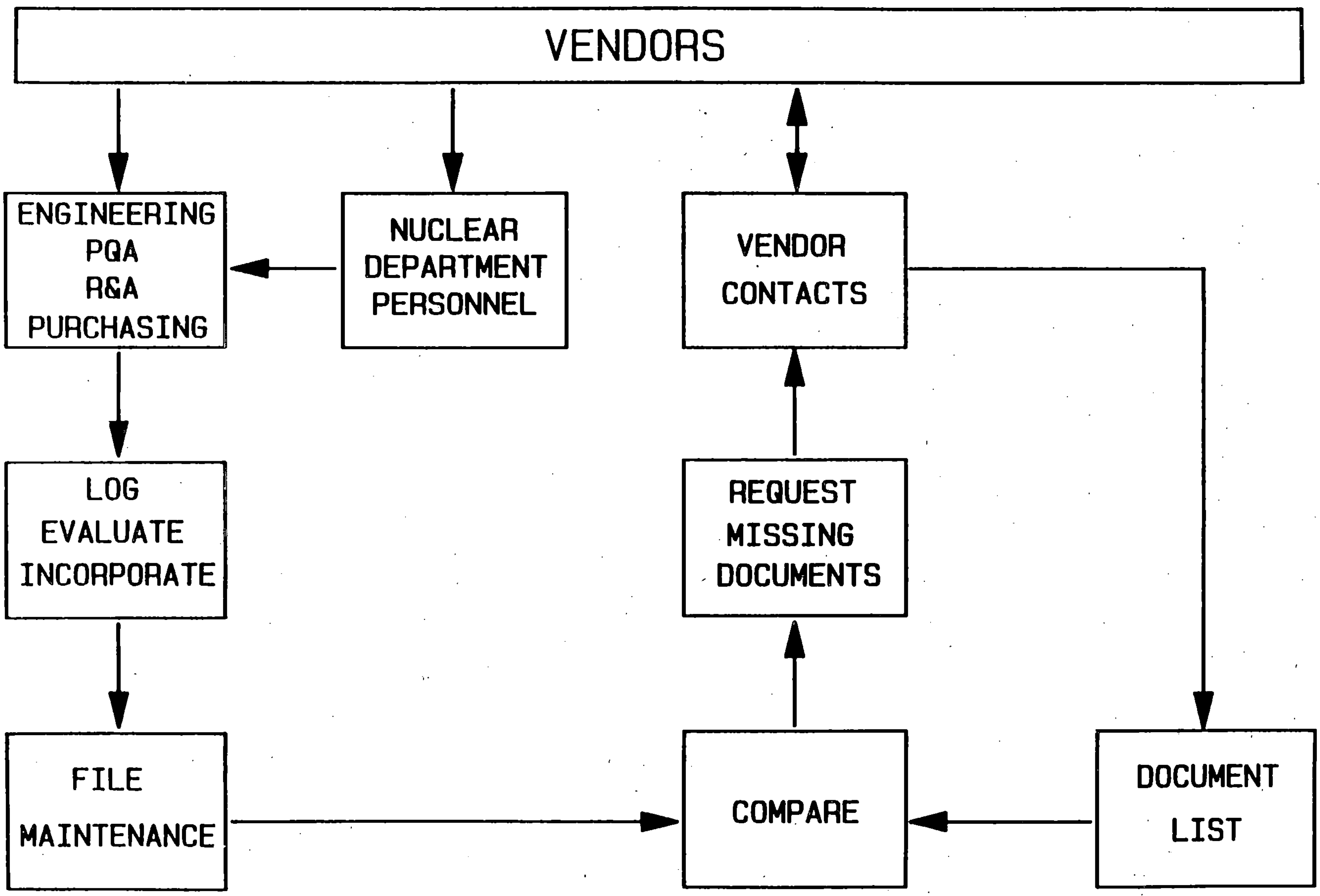
VENDOR INFORMATION CONTROL

- NRC FINDING
  - FAILURE TO REVIEW, EVALUATE, INCORPORATE AND MAINTAIN CERTAIN VENDOR TECHNICAL INFORMATION

- DESCRIPTION OF DEFICIENCY
  - DURING INSPECTION PSE&G WAS UNABLE TO RETRIEVE THREE REQUESTED DOCUMENTS:
    - ▲ LIMITORQUE MAINTENANCE UPDATE 89-1
    - ▲ LIMITORQUE MAINTENANCE UPDATE 90-1
    - ▲ MOVATS ENGINEERING REPORT 5.0

- PSE&G RESPONSE
  - LIMITORQUE MAINTENANCE UPDATES SHOULD HAVE BEEN RETRIEVABLE
  - LACK OF RETRIEVABILITY OF MOVATS ENGINEERING REPORT DOES NOT SUPPORT FINDING

PSE&G VENDOR INFORMATION CONTROL PROGRAM



91EC1-36

MOV PROGRAM

VENDOR INFORMATION CONTROL

EVALUATION OF DEFICIENCY - MOVATS DOCUMENTS

- MOVATS ENGINEERING REPORT 5.0
  - MOVATS ISSUED REV. 0 REPORT INTERNALLY ON 1/3/91
  - PSE&G BECAME AWARE DURING TEAM INSPECTION ON 7/16/91
  - REVIEW OF MOVATS PROCESS
    - ▲ ENGINEERING REPORTS NOT ISSUED DIRECTLY TO INDUSTRY
    - ▲ VENDOR INFORMATION ISSUED UNDER TECHNICAL NOTICES
  - MOVATS ENGINEERING REPORT 5.0 WAS NOT INTENDED FOR INDUSTRY USE
  - ABSENCE OF ENGINEERING REPORT NOT INDICATIVE OF PROGRAM DEFICIENCY

- MOVATS TECHNICAL NOTICES
  - MOVATS ISSUED 10 TECHNICAL NOTICES BETWEEN 1988 AND 1990 NOT RETRIEVABLE BY PSE&G
    - ▲ RECEIVED BY INDIVIDUALS AT PSE&G
    - ▲ FAILED TO FOLLOW PROCEDURE TO INPROCESS INTO PSE&G SYSTEM

MOV PROGRAM

VENDOR INFORMATION CONTROL

EVALUATION OF DEFICIENCY - LIMITORQUE DOCUMENTS

- LIMITORQUE MAINTENANCE UPDATES 89-1, 90-1
  - LIMITORQUE MAINTENANCE UPDATE 89-1 ISSUED 12/89
    - ▲ RECEIVED BY INDIVIDUAL AT PSE&G (1/90)
    - ▲ FAILED TO FOLLOW PROCEDURE TO INPROCESS INTO PSE&G SYSTEM
  - LIMITORQUE MAINTENANCE UPDATE 90-1 ISSUED 5/90
    - ▲ LIMITORQUE UNABLE TO PRODUCE MAILING LIST
    - ▲ NO PSE&G PERSONNEL ACKNOWLEDGED RECEIPT
  - MUG MEETING PARTICIPATION 7/90
    - ▲ PSE&G PERSONNEL REQUESTED AND RECEIVED ALL LIMITORQUE MAINTENANCE UPDATES
    - ▲ FAILED TO FOLLOW PROCEDURE TO INPROCESS INTO PSE&G SYSTEM
- LIMITORQUE MAINTENANCE UPDATE 88-1
  - ISSUED BY LIMITORQUE 8/88 BUT NOT RETRIEVABLE BY PSE&G
    - ▲ TRANSMITTED TO INDIVIDUALS AT PSE&G
    - ▲ NO PSE&G PERSONNEL ACKNOWLEDGED RECEIPT

MOV PROGRAM

VENDOR INFORMATION CONTROL

SAFETY SIGNIFICANCE

- MOVATS TNs 88-01 THROUGH 04, 89-01 THROUGH 04 AND 90-01 (TEN TOTAL INCLUDING SUPPLEMENTS)
  - ENGINEERING REVIEW INDICATES NO SAFETY SIGNIFICANCE ISSUES
  - APPLICABLE CONTENT TO BE INCORPORATED INTO THE MOV PROGRAM

- LIMITORQUE MUs 88-1, 90-1
  - ENGINEERING REVIEW INDICATES NO SAFETY SIGNIFICANCE ISSUES
  - APPLICABLE CONTENT TO BE INCORPORATED INTO THE MOV PROGRAM

- LIMITORQUE MU 89-1
  - EVALUATION OF 13 HOPE CREEK VALVES FOR WHICH LIMITER PLATES WERE REMOVED INDICATED NO SAFETY SIGNIFICANCE

VENDOR INFORMATION CONTROL

ROOT CAUSE ANALYSIS

- LACK OF UNDERSTANDING BY LIMITORQUE AND MOVATS OF PSE&G POINTS OF CONTACT FOR CORRESPONDENCE

- INADEQUATE TRAINING
  - LESS THAN ADEQUATE UNDERSTANDING BY SOME PSE&G PERSONNEL OF ELEMENTS OF VENDOR DOCUMENT CONTROL PROGRAM

MOV PROGRAM

VENDOR INFORMATION CONTROL

RECENT PROCESS IMPROVEMENTS

- ISSUED NUCLEAR DEPARTMENT ADMINISTRATIVE PROCEDURE TO CLARIFY THE OPERATING EXPERIENCE FEEDBACK PROGRAM (3/90)
- PUBLISHED POLICY TO CLARIFY REQUIREMENTS FOR INPROCESSING OF VENDOR TECHNICAL DOCUMENTS RECEIVED BY PSE&G INDIVIDUALS (5/90)
- ESTABLISHED PROGRAMMATIC STANDARD TO FORMALIZE VENDOR CONTACT PROGRAM (12/90)
- UPDATED IMPLEMENTING PROCEDURE TO ALIGN TRANSMITTAL AND INPROCESSING OF VENDOR TECHNICAL DOCUMENTS WITH PSE&G POLICY (7/91)
- ISSUED LETTERS TO VENDORS (3/91 - 7/91) TO:
  - REITERATE PSE&G INPROCESSING REQUIREMENTS
  - ESTABLISH AND MAINTAIN DOCUMENT TRANSMITTAL LISTINGS

MOV PROGRAM

VENDOR INFORMATION CONTROL

CORRECTIVE ACTIONS

- INPROCESSED DOCUMENTS IDENTIFIED AS MISSING INTO VENDOR TECHNICAL DOCUMENT PROGRAM (8/16/91)

- HAD LIMITORQUE AND MOVATS MODIFY THEIR MAILING LISTS TO CONFORM TO VENDOR CONTACT PROGRAM (8/23/91)

- COMPLETED SAMPLING OF OTHER VENDORS TO ASSURE RECEIPT OF ISSUED DOCUMENTS (9/5/91)

- ISSUED LETTER UNDER VICE PRESIDENT SIGNATURE TO ALL PSE&G PERSONNEL REITERATING IMPORTANCE OF AND THEIR RESPONSIBILITY TO VENDOR DOCUMENT CONTROL POLICY, PROGRAM AND PROCEDURES (9/6/91)

MOV PROGRAM

VENDOR INFORMATION CONTROL

CORRECTIVE ACTIONS (cont)

- WILL REVIEW VENDOR DOCUMENT CONTROL PROCESS TO DETERMINE POSSIBLE ADDITIONAL IMPROVEMENTS (11/29/91)

- WILL FORMALIZE PROCESS FOR VALIDATION AND RECONCILIATION OF LISTS FROM VENDOR CONTACT PROGRAM (11/29/91)

- PERFORMING COMPARISON OF PSE&G PROGRAM TO THOSE OF UTILITIES RECOGNIZED BY INPO FOR THEIR VENDOR INFORMATION PROGRAMS (11/29/91)

- EVALUATE ADDITIONAL TRAINING NEEDS AND IMPLEMENT ADDITIONAL TRAINING AS REQUIRED

- WILL PROVIDE OVERSIGHT OF THE VENDOR CONTACT PROGRAM THROUGH USE OF QA VENDOR AUDITS/SURVEILLANCES

- ON PERIODIC BASIS WILL PERFORM AN EFFECTIVENESS REVIEW OF VENDOR TECHNICAL DOCUMENT PROGRAM IN ADDITION TO NORMAL QA AUDITS OF THE PROGRAM

MOV PROGRAM

VENDOR INFORMATION CONTROL

SUMMARY

- PSE&G UNABLE TO RETRIEVE DOCUMENTS THAT SHOULD HAVE BEEN RETRIEVABLE
- PERFORMED ASSESSMENT OF THIS SITUATION
- MADE RECENT PROCESS IMPROVEMENTS, TOOK IMMEDIATE CORRECTIVE ACTION AND COMMITTED TO LONG-TERM CORRECTIVE ACTIONS
- VENDOR INFORMATION CONTROL COMPLEX ISSUE
- RECOGNIZED NEED FOR CONTINUAL IMPROVEMENT
- COMMITTED TO WORK WITH VENDORS AND INDUSTRY TO CONTINUE FURTHER IMPROVEMENT

MOV PROGRAM

PSE&G ASSESSMENT OF POTENTIAL VIOLATIONS

GL 89-10, SUPPLEMENT 3 RESPONSE

- PSE&G BELIEVES THAT A 10CFR50.9 VIOLATION IS NOT APPLICABLE FOR THE FOLLOWING REASONS:
  - A PSE&G REVIEW OF SEVERAL EVALUATIONS WAS PERFORMED THAT PROVIDED A SOUND TECHNICAL BASIS FOR DATA IN SUPP 3
  - INSPECTION REPORT ISSUES IDENTIFIED WERE ADDRESSED BASED ON BEST TECHNICAL INFORMATION AVAILABLE
  - DIFFERING PROFESSIONAL OPINIONS EXIST ON SOME TECHNICAL ISSUES
  - FAILED TO ADEQUATELY COMMUNICATE THE LOGIC FOR OUR CONCLUSION OF OPERABILITY BOTH IN OUR SUPPLEMENT 3 RESPONSE AND DURING THE INSPECTION
  - INFORMATION PROVIDED WAS NOT INACCURATE OR INCOMPLETE
  - NO SAFETY SIGNIFICANCE RELATED TO SUPP 3 ISSUE
  - ACTIONS UNDERWAY TO IMPROVE EFFECTIVENESS OF COMMUNICATIONS WITH THE NRC
  - PAST PERFORMANCE RELATIVE TO WRITTEN COMMUNICATION/RESPONSE TO NRC ISSUES HAS BEEN GOOD

MOV PROGRAM

PSE&G ASSESSMENT OF POTENTIAL VIOLATION

TORQUE SWITCH SETTING AND VENDOR DOCUMENT CONTROL ISSUES

- PSE&G DOES NOT DISPUTE THE VIOLATIONS
- SEVERAL MITIGATING FACTORS APPLY:
  - COMPREHENSIVE CORRECTIVE ACTIONS TAKEN/UNDERWAY
  - DEMONSTRATED NO SAFETY SIGNIFICANCE
  - CONTINUING FOCUS ON IMPROVING SUBJECT PROGRAMMATIC AREAS
  - PAST PERFORMANCE ON IDENTIFICATION AND CORRECTIVE ACTIONS OF DEFICIENCIES HAS BEEN GOOD
- PSE&G BELIEVES THAT ESCALATED ENFORCEMENT SHOULD NOT BE APPLIED TO THESE ISSUES

MOV PROGRAM

SUMMARY

- NO SAFETY ISSUE IDENTIFIED WITH OPERABILITY OF THE MOVs
- PSE&G RECOGNIZES THAT MORE MANAGEMENT ATTENTION/OVERSIGHT IS NEEDED FOR MOV PROGRAM
- FULLY PLANNED PROJECT IN PLACE
  - PROGRAM DESCRIPTION BY 10/31/91
  - PROGRAM COMPLETION BY 6/94 (ORIGINAL COMMITMENT)
- FAILED TO ADEQUATELY COMMUNICATE THE LOGIC FOR OUR CONCLUSION OF OPERABILITY IN OUR SUPPLEMEN 3 RESPONSE
- DATA PROVIDED WAS SUPPORTED BY REVIEW OF SEVERAL EVALUATIONS/BEST TECHNICAL DATA AVAILABLE
- ROOT CAUSES DETERMINED FOR TORQUE SWITCH/VENDO DOCUMENT ISSUES
- COMPREHENSIVE CORRECTIVE ACTION TAKEN/CONTINUING
- ESCALATED ENFORCEMENT NOT WARRANTED

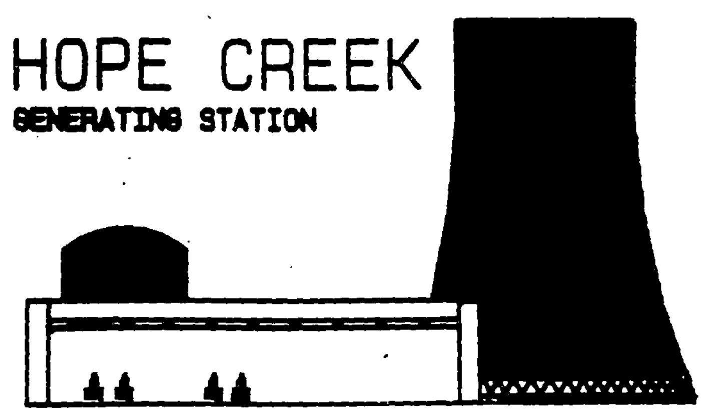
PSE&G *Public Service Electric and Gas Company*

ATTACHMENT 3

NRC ENFORCEMENT CONFERENCE

MOV PROGRAM

SEPTEMBER 9, 1991



NRC ENFORCEMENT CONFERENCE
SEPTEMBER 9, 1991

HANDOUT TABLE OF CONTENTS

INTRODUCTION

The following provides a written discussion of topics discussed at the September 9, 1991 Enforcement Conference. In addition, issues covered in the inspection report, but not addressed at the enforcement conference are discussed.

OVERALL SAFETY ASSESSMENT

A detailed review of our Supplement 3 analysis has verified that the six identified valves will close under design basis accident conditions.

This conclusion is based on the fact that substantial margins exist in all cases due to torque switch bypass circuitry. The bypass circuitry assures that adequate thrust is available to overcome potential valve factor concerns and to achieve disk-to-seat overlap. The structural capability of the valve assemblies are adequate to withstand the stresses associated with torque switch bypass circuitry. Subsequent evaluations, as well as a review of data available prior to our Supplement 3 response, have confirmed our operability conclusions.

MOV PROGRAM SCHEDULE

Phase I of our Generic Letter 89-10 program was originally scheduled for implementation between December 1989 and October 1990 while Phase II was scheduled to begin with the completion of Phase I in October 1990 and to continue through June 1994. Due to unplanned burdens on our valve engineering resources, which were not adequately monitored and managed, Phase I was not completed until June 1991. Although the initiation of our Phase II was delayed until June 1991, PSE&G management is committed to take whatever actions are necessary to complete Phase II in accordance with the original schedule (June 1994).

The schedule for completion of our program description has slipped ten months from January 1, 1991 to October 31, 1991.

PROGRAM DESCRIPTION

Phase I Activities

Phase I of our Generic Letter 89-10 program included four major activities. The first activity involved development of engineering requirements and program implementation plans. The second activity consisted of initiating a pilot program by performing a detailed analysis for 15 selected valves (9 Salem valves and 6 Hope Creek valves). As a result of the Phase I delay, we were aware of concerns with the six valves which were addressed by Generic Letter 89-10, Supplement 3 prior to performing the detailed analysis. The six Supplement 3 valves were therefore the six Hope Creek valves evaluated during Phase I. The third activity involved assessing our existing MOV program against the Generic Letter 89-10 recommendations. This activity was intended to identify any necessary improvements to our existing program. The final major activity consisted of identifying scope items for inclusion in Phase II of the program.

Lessons Learned - Phase I

As a result of Lessons Learned during Phase I, numerous improvements were identified for inclusion in Phase II. The principal improvements included enhancement of the current nuclear department policy governing the overall MOV program, development of programmatic standards for control of future MOV activities, consolidation and reconciliation of existing data sources, and development of improved methods and procedures for MOV testing and maintenance.

Phase II Activities

Phase II includes two major activities. The first activity consists of implementing policy and formal program requirements to meet Generic Letter 89-10 objectives. This activity includes developing program implementing procedures which cover management policy, data collection/reconciliation, functional evaluations, corrective action, and trending. Development of improved maintenance methods for maintenance and testing are also contained within this activity. The second activity involves establishing detailed schedules for program development and implementation. Our programmatic standard along with its twelve appendices constitute our MOV Program description and will be completed and approved by October 31, 1991. A project plan, which outlines organizational and functional responsibilities relative to our MOV program, has also been prepared. In addition to the development and processing of the programmatic standard, its appendices, and our project plan, current ongoing activities include data gathering and MOV evaluations to support maintenance activities. Phase II is scheduled to be completed by June 1994.

PROGRAM DESCRIPTION DEVIATION

ISSUE

"In a submittal dated August 31, 1990, to the NRC, the licensee stated that a detailed GL 89-10 program description will be available onsite on January 1, 1991. However, at the time of the inspection, the licensee had not established an approved program description."

PSE&G RESPONSE

We agree with the deviation as stated.

ROOT CAUSE

One root cause of the deviation a was failure to recognize and address the over-commitments on our valve engineering resources in a timely manner. The over-commitments resulted in a ten month slip in schedule primarily due to turnover of key personnel and diversion of resources to other significant issues. Other significant issues included ECCS and MSIV concerns at Salem which resulted in self-imposed shutdowns during the Spring, Summer, and Fall of 1990. A second root cause was failure to utilize our monitoring and control process for tracking Generic Letter 89-10 program description commitments. Had this commitment been properly tracked, an alternative means of alerting management to the approaching commitment would have existed.

CORRECTIVE ACTIONS

The Vice President - Nuclear Engineering has counseled all management personnel directly involved.

A letter has been sent from the Vice President - Nuclear Engineering to all Nuclear Engineering Department personnel communicating his expectations relative to control of commitments, schedule adherence, and timely reconciliation of resource issues and requesting managers to review all regulatory programs to ensure similar problems do not exist in other areas.

An independent review will be conducted to assess the effectiveness of our regulatory commitment tracking process. This review will be completed by December 31, 1991.

The project plan was re-established in October 1990. Phase I activities are now complete, Phase II activities are underway, and final completion is expected by June 1994.

SUPPLEMENT 3 RESPONSE

PSE&G SUMMARY RESPONSE TO 10CFR.50.9 CONCERN

Although the technical basis for our conclusions was not adequately communicated, our response to Supplement 3 was complete and accurate based on engineering judgement and evaluation. Although thrust values were taken from valve data sheets, these values were considered to be the best available design information. Our operability conclusion was based on margins available considering torque switch bypass circuitry. All available data and evaluations were reconciled prior to our Supplement 3 response. Although the basis for our operability conclusion was not clearly communicated, an adequate and documented basis existed prior to our Supplement 3 response. We do not believe that our Supplement 3 response was either inaccurate or incomplete, and therefore, a 10CFR50.9 violation is not justified.

Each of the individual findings relative to the potential 10CFR50.9 violation is discussed below. The first two findings have significance relative to 10CFR50.9, however, although we consider the other three inspection findings to be important issues, they do not relate to any issue specifically addressed in Supplement 3. Although they are technical issues targeted for resolution under GL 89-10, these issues were not required to be resolved prior to our Supplement 3 response. At the time of our Supplement 3 response, we were not in a position to accelerate our program to address the issues covered under the final three findings. Since acceleration of our program for the subject valves was a suggestion and not a requirement, we chose to make use of the best information available at the time. The issues described in the final three findings will be addressed in Phase II of our program.

SPECIFIC FINDINGS

FINDING

"Thrust values provided were not derived from detailed evaluations, but were obtained from valve data sheets. Some of this data was provided by the valve manufacturer (Anchor Darling) as part of the original plant design in the later 1970's."

"The licensee had three sets of evaluations (dated December, 1990; February, 1991; and March, 1991;) performed by two licensee contractors indicating that the thrust requirements provided in the response might be nonconservative. However, the existence of this data was not mentioned in the licensee's March 8, 1991 letter to the NRC."

SUPPLEMENT 3 RESPONSE

PSE&G POSITION

We agree that thrust values provided in our Supplement 3 response were obtained from valve data sheets. Our operability conclusion was based on our evaluation and reconciliation of all available information and our determination that positive margin existed for all valves when credit was taken for torque switch bypass circuitry. The valve data sheet values represented the best available design information and were, therefore, the thrust values provided in our Supplement 3 response.

Prior to submittal of our Supplement 3 response, the following reviews were performed by PSE&G. The valve data sheet thrust values were reviewed and determined to be conservatively derived. The Phase I results which consisted of vendor evaluations were reviewed and found to be conservative. All available data and evaluations, including the three sets of evaluations noted in the NRC finding, were reconciled, and the values from the valve data sheets were determined to be the best available design information. The reconciliation and review of available information was documented in the draft engineering response to Supplement 3. The draft engineering response demonstrated that, with credit taken for torque switch bypass circuitry, substantial margins exist for all of the subject valves irrespective of which of the various required thrust values were used. The torque switch bypass circuitry allows thrust up to motor rated torque which greatly exceeds torque switch trip thrust. The draft engineering response was overlooked and not provided to NRC inspectors during the inspection.

Based on the above, we conclude that, although the basis for our operability conclusion was not clearly communicated, an adequate and documented basis existed prior to our Supplement 3 response to conclude that the valves were capable of performing their intended safety function under design basis accident conditions. Information we considered material to drawing this conclusion was contained in our Supplement 3 response.

SUPPLEMENT 3 RESPONSE

FINDING

"The licensee did not have any technical justification for their use of a non-conservative 0.3 disk factor and 0.15 stem friction factor in determining their required and available thrust capabilities."

PSE&G POSITION

Current disk and stem friction factors are the result of best available vendor recommendations. Differing professional opinions exist within the industry relative to specific values to be used for disk factor and stem friction factor. Our Phase II program will re-evaluate valve factors based on industry consensus. We have historically included a margin of 15% ± 5% in our MOV maintenance procedures to address engineering uncertainties and instrument inaccuracies. Torque switch bypass allows application of full rate motor torque which allows accommodation of higher valve factors.

FINDING

"The thrust margins shown in the response did not include the effects of diagnostic instrument inaccuracies or the rate of loading effects."

PSE&G POSITION

Reported margins were not adjusted for instrument inaccuracies. A margin of 15% ± 5% has been historically included in our MOV maintenance procedures to address engineering uncertainties and instrument inaccuracies. The margins reported in our response are in fact the allowance for engineering uncertainties. The actual margins would be greater based on the higher thrust which could be developed when the torque switch is bypassed. Additionally, rate of loading is considered to be a concern only when there is an interface between the torque switch and spring pack. As such, rate of loading is not a concern over the majority of the valve stroke due to the bypass circuit. Final disposition of these issues will be accomplished during Phase II of our MOV program.

SUPPLEMENT 3 RESPONSE

FINDING

"The licensee obtained the thrust values from test whose conditions were significantly different from those during design basis accident conditions. Specifically, the licensee provided the thrust values when the torque switches tripped during the test. However, during an accident, the torque switches will be bypassed and the available thrust will be dependent entirely on motor thrust capability."

PSE&G POSITION

The reported thrust values reflected the most recent diagnostic results obtained under static conditions and were based on torque switch trip. The bypass circuitry allows thrust up to motor rated capability. Motor rated torque exceeds torque switch trip thrust and therefore torque switch trip thrust values are conservative. It is noted that defeat of the bypass circuitry occurs after disk-to-seat overlap is accomplished. Differential testing will be performed under Phase II of our program where practical.

ISSUE

"However, the licensee still has to perform detailed evaluations to determine the margin of safety available after considering such factors as instrument accuracies and rate of loading."

PSE&G POSITION

Instrument accuracies and rate of loading were previously addressed. Based on the information provided, reasonable assurance of operability existed due to bypass circuitry even when instrument uncertainty is considered. Final disposition of these issues will be accomplished during Phase II.

SUPPLEMENT 3 RESPONSE

ACTIONS TAKEN TO DATE AND ACTIONS TO BE COMPLETED

We are sensitive to the significance of these issues and the implications of our less than clear communication. As such, we have taken actions to confirm our previous conclusions and investigated the circumstances leading to the issuance of our response. Additionally, we have identified follow-up activities to avoid similar occurrences in the future. Identified actions are as follows:

Actions Taken To Date

> We have re-evaluated existing data to confirm our original operability conclusions. In addition an independent assessment has been conducted by our Nuclear Safety Review Group. The Vice President Nuclear Operations has discussed the issue with E&PB management and reiterated his expectations relative to communications with the NRC.

Actions To Be Completed

> A letter will be issued to all Nuclear Department Personnel to emphasize our obligation relative to clear and complete information in our communications with the NRC. Training will be conducted within the Licensing Department to ensure that expectations and responsibilities are understood relative to communication with the NRC. Our Supplement 3 response will be revised by September 30, 1991.

SUMMARY

Our response to Generic Letter 89-10, Supplement 3 was accurate and complete based on engineering judgement and evaluation and therefore, we are not in violation of 10CFR50.9.

Technical concerns raised in Generic Letter 89-10 will be addressed under Phase II of our program. Technical concerns to be addressed include diagnostic equipment inaccuracies, rate of loading, and conservative valve factors.

Substantial margins exist to assure proper valve function under design basis accident conditions.

PSE&G is actively involved in industry groups following MOV issues including MUG, BWROG, and EPRI.

Although we failed to adequately communicate the technical basis for our conclusions, there was no intention to mislead the NRC.

Our Supplement 3 response will be resubmitted to clarify the logic for our operability determination.

TORQUE SWITCH SETTING

The finding, as described in the inspection report, states - modification of MOV torque switch settings without a documented engineering or safety evaluation. Specifically, the limiter plates were removed from 2 RCIC valve motor operators. These plates were removed so that the switch settings could be increased to obtain the thrust required. No engineering evaluation had been documented. PSE&G agrees with this finding.

Our understanding of the issue, back during startup of Hope Creek, included the following elements:

- The switch setting was intended to provide the design basis seating thrust for the valve;

- The placement of the limiter plate was based on the vendor's calculation; this methodology (calculation and subsequent placement) was less accurate than direct measurement would yield.

- The switch settings were administratively controlled by the maintenance procedure; namely, the approval of the system engineer was required, and verbal occurrence from Limitorque and the valve manufacturer was obtained prior to the removal of a limiter plate.

However, we did not document the evaluation process or the telephoned concurrences. We conclude that the root cause was that the past procedural controls failed to adequately address documentation requirements.

We have revised our maintenance procedures to require that a Deficiency Report (DR) be initiated for any future removal of limiter plates. DRs require the 50.59 process, thereby ensuring the proper documentation of such evaluations.

We have done a review of post-startup maintenance records to identify similar occurrences. To date, we have identified 13 valves with limiter plates removed. These have been written up on DEFs (Discrepancy Evaluation Forms) and are being evaluated under the Engineering Discrepancy Evaluation process. This process starts with an initial operability screening, next, the DEF is prioritized based on its safety significance (PRA), and then evaluated in light of Limitorque Maintenance Update 89-1. Our screening and evaluation shows there is no safety significance for the removal of limiter plates on the 13 Hope Creek valves.

We will complete our review of all pre-startup documentation by November 15, 1991.

Again, our present program for the control of torque switch settings requires the initiation of a deficiency report in order to exceed specified settings. The deficiency report will document the engineering evaluation and ensure that 50.59 is addressed.

In summary:

- Our past evaluation were not formally documented;
- Our review has shown no significant safety concerns;
- The records at the Salem station are also being reviewed for similar occurrences;
- Any such discrepancies noted will also be evaluated using the DEF process; and
- We have revised our procedures to ensure that any future adjustments will be documented.

VENDOR INFORMATION CONTROL ISSUE

NRC Finding

Failure to review, evaluate, incorporate and maintain vendor technical information is a potential violation of 10CFR50, Appendix B, Criterion V.

Description of Deficiency

During Inspection No. 50-354/91-80, PSE&G was unable to retrieve three (3) documents requested by the inspection team:

- Limitorque Maintenance Update (MU) 89-1, "Maximum Torque Switch Settings and Other Issues" (Dated 10/89)

- Limitorque Maintenance Update (MU) 90-1, "Hydraulic Lock and Torque Spring Assembly Relaxation" (Dated 5/90)

- MOVATS Engineering Report (ER) 5.0, "Equipment Accuracy Summary" (Dated 1/3/91)

PSE&G Position

Limitorque MUs 89-1 and 90-1 should have been retrievable through our Vendor Document Control Programs at the time of the inspection.

Lack of retrievability of MOVATS ER 5.0 does not support the NRC finding and thus is not a basis for a violation. PSE&G will elaborate on this position later in this document under "Evaluation of Deficiency".

Vendor Information Control Program

PSE&G is extremely sensitive to the issues involved with Vendor Information Control. It is a complex issue requiring diligence on our part as we coordinate with many (approximately 300) vendors and process a multitude (approximately 7500) documents per year.

The processes we have are in accordance with INPO Good Practice and have been embellished over the years to ensure the integrity of our programs.

The PSE&G Vendor Information Control Program is composed of four (4) primary elements each of which have associated process procedures addressing the logging, evaluation, incorporation and maintenance of certain Vendor Information.

These elements were established along the organizational lines of responsibility of our Engineering, Procurement Quality Assurance, Reliability and Assessment; and Purchasing Organizations.

The four (4) primary elements are supplemented by two secondary elements in order to maximize the integrity of our program. These secondary elements are our Vendor Contact Program; and the requirement to and provision for individual Nuclear Department personnel receiving Vendor Information to in-process same into our Vendor Information Control Program.

There has been continuing improvement/modifications made to these programs over the last few years. Additional improvements that we will be making as a result of the inspection findings are discussed in the following sections.

Evaluation of Deficiency

1. MOVATS Engineering Report 5.0

- MOVATS issued ER 5.0, Revision 0 internally on 1/3/91. PSE&G first became aware of ER 5.0's existence during the inspection on 7/16/91 when requested to retrieve it by the inspection team.

- A subsequent review of the MOVATS process indicated that:

  - ERs such as ER 5.0 are internal MOVATS documents and, as such, are not directly transmitted to the industry (including PSE&G).

  - Relevant information resulting from ERs would have been issued to the industry by MOVATS in the form of Technical Notices.

- MOVATS has not yet incorporated the results of ER 5.0 into a Technical Notice (TN):

  - The report was issued internally for the purpose of consolidating data and to be used as a training aid.

  - PSE&G is aware that ER 5.0 involves instrument accuracies associated with MOVATS equipment which is currently an unresolved industry issue. PSE&G personnel actively participate in the Motor

Operated Valve User's Group (MUG) and are closely following this issue to resolution.

- PSE&G is in receipt of, and in the process of reviewing, a letter from Westinghouse ITI MOVATS dated August 16, 1991 pertaining to this issue.

- Based on the preceding discussion, the absence of ER 5.0 from our Vendor Document Control System is not indicative of a deficiency in our program.

2. MOVATS Technical Notices (TNs)

- Further evaluation of documents issued by MOVATS indicated that the following ten (10) TNs had been distributed but were not retrievable through PSE&G's Document Control System:

TN 88-01 "Differential Pressure Thrust Calculations"

TN 88-02 "Spring Pack Response to Stem Loads"

TN 88-03 "Use of the AC Motor Load Unit"

TN 88-04 "2151 Mainframe"

TN 89-01 "Locked Rotor Condition Due to Grease Relief Kit"

TN 89-02 "Spring Pack Response Under Differential Pressure"

TN 89-03 "Use of AC Motor Load Unit"

TN 89-03 Supplement 1 "Use of AC Motor Load Unit"

TN 89-04 "Output Imbalance in HBC Gearboxes and Use of the BART System for Measuring Gearbox Output Torque"

TN 90-01 "Rate of Loading"

- These TNs were transmitted directly to individual PSE&G personnel who had participated in the MOVATS training program as evidenced by the MOVATS mailing list.

- None of the acknowledged PSE&G recipients of the subject TNs had attempted to in-process them into our Document Controls system insofar as they were perceived to be personal copies.

3. Limitorque Maintenance Updates 89-1 and 90-1

- On 12/22/89, Limitorque issued MU 89-1.

  - MU 89-1 was directly transmitted to individual PSE&G personnel that participate in the MUG as evidenced by the Limitorque Mailing List.

- In January 1990, MU 89-1 was received by a Maintenance Engineer at Hope Creek for information under a Limitorque cover letter dated 12/22/89.

  - No further action was taken insofar as it was perceived as a personal copy and assumed to be in our system.

- In May 1990, Limitorque apparently transmitted MU 90-1

  - Limitorque can not provide evidence (i.e., Mailing List) that it was transmitted directly to PSE&G.

  - No PSE&G personnel known to be on Limitorque's recent mailing list acknowledge receipt of MU 90-1.

- In July 1990, the MUG held their summer meeting

  - Salem System Engineer obtained copies of all MUs issued to date directly from Limitorque after they were identified at the MUG meeting.

  - He was not aware of the procedural guidance with regard to the in-processing of vendor documents obtained in this manner.

4. Limitorque Maintenance Update 88-1

- Further evaluation of documents issued by Limitorque indicated that one (1) additional Maintenance Update, MU 88-1, had been distributed but was not retrievable through PSE&G's Document Control Program.

- MU 88-1 had been transmitted directly to two (2) PSE&G personnel who had been involved in valve maintenance activities at the time of issuance as evidenced by Limitorque's mailing list.

  - One PSE&G recipient left PSE&G's employment shortly after MU 88-1 was issued.

  - The other individual, who is no longer involved with valve maintenance, could not acknowledge receipt of MU 88-1.

<u>Safety Significance</u>

MOVATS Technical Notices

- TN 88-01 through 88-04, 89-01 through 89-04, and 90-01

  - All ten (10) MOVATS Technical Notices are undergoing the required Engineering review in accordance with PSE&G's process. A preliminary review of the content indicates that there are no safety significant issues involved.

  - All applicable information that is included in these Technical Notices will be factored into PSE&G's MOV program.

Limitorque Maintenance Updates

- MU 88-1 and MU 90-1

  - Both Limitorque Maintenance Updates are undergoing the required Engineering review in accordance with PSE&G's process. A preliminary review of the content indicates that there are no safety significant issues involved.

  - All applicable information that is included in these Maintenance Updates will be factored into PSE&G's MOV program.

- MU 89-1

  - Evaluation of 13 Hope Creek valves for which Limiter Plates were removed indicated no safety significance.

Survey of Other Vendors

PSE&G contacted a sample (10%) of our vendors and requested them to identify updated documents that they have provided to PSE&G in the last two (2) years.

The vendors identified a total of 57 documents that had been transmitted during the 2 year period.

PSE&G was able to retrieve 100% of the documents through our Vendor Document Control System.

PSE&G therefore concludes that: 1) this issue is limited to MOVATS and Limitorque, and 2) resulted from the direct interface between the two vendors and the individual PSE&G personnel involved with MOV maintenance issues.

Root Cause Analysis

- Lack of clear understanding by Limitorque and MOVATS of the PSE&G Vendor Document Control Program and their responsibilities regarding points-of-contact for correspondence.

- Inadequate training was a major contributing factor as evidenced by the unfamiliarity on the part of some PSE&G personnel with "elements" of the Vendor Document Control program. Specifically, personnel not normally in the "mainstream" of the Vendor Document Control Programs did not understand their responsibility regarding the in-processing of documents received directly from vendors.

Recent Process Improvements

- PSE&G has, and will continue to recognize that vendor information control is a very complex and extremely important issue which must be thoroughly and continuously addressed in order to operate our nuclear units safely and reliably.

- To this end, PSE&G is committed to work with our vendors and the industry to demonstrate continued improvement in this area to our customers, regulators, and ourselves.

- On 3/7/90, PSE&G issued Procedure NC.NA-AP.ZZ-0054(Q) Revision 0, "Operating Experience Feedback (OEF) Program", which clarified the Operating Experience Feedback Program.

- On 5/31/90, PSE&G issued Procedure VPN-EDP-01 Revision 1, "Vendor Document Control", which provides the policy for our Vendor Technical Document (VTD) Program. This revision included the addition of Hope Creek Station applicability throughout the text, and clarified the requirements for receiving, reviewing and distributing VTDs.

- On 12/31/90, Programmatic Standard DE-PS.ZZ-0031(Q) Revision 0, "Vendor Contact Program", was issued which formalized our vendor contact program as required by GL 90-03.

- On 3/8/91 and 6/12/91, PSE&G issued letters to vendors for the purpose of clarifying document transmittal requirements and PSE&G contacts (Engineering, Procurement Quality Assurance, Reliability and Assessment; and purchasing); and establishing a document tracking system with each vendor.

- On 7/29/91, Procedure NC.DE-AP.ZZ-0006(Q) Revision 3, "Vendor Document Control Program", was issued. In part, this revision aligned the responsibilities of Nuclear Department organizations with those outlined in procedure VPN-EDP-01.

- On 7/30/91, PSE&G issued follow-up letters requesting a response by 8/22/91 to all Suppliers With Approved Quality Systems (SWAQS) lists and GL 90-03 "Category b" vendors who had not responded to our previous letters (this included Limitorque and MOVATS).

Corrective Actions

- The following corrective actions have been completed:

  - All documents requested by the NRC during the the inspection which were not retrievable through our system have since been obtained, in-processed to our Vendor Document Control System and are currently undergoing review.

  - Limitorque and MOVATS, as directed by PSE&G, has modified their mailing list to conform with our Vendor Contact Program.

  - A sampling of other vendors to assure receipt of all documents issued to PSE&G has been completed.

- A letter has been issued, under Vice Presidential signature, to all PSE&G Nuclear Department Personnel reiterating the importance of and their responsibility to the Vendor Document Control Policy, Program and Procedures.

- The following corrective actions will be completed by November 29, 1991:

  - All Vendor Document Control Program Procedures will be reviewed to ensure clarity and determine additional areas for potential improvement. As part of this review, PSE&G will address procedural consistency and identification of proper responsibilities, authorities and interfaces.

  - The process for validating and reconciling vendor-supplied document distribution lists will be formalized; this will provide confirmation upon receipt of documents transmitted by vendors.

  - An evaluation will be conducted to compare PSE&G's programs to those of other utilities that have been recognized by INPO as having excellent Vendor Information Control Programs.

- The following on-going corrective actions will be implemented after completion of the programmatic changes described above:

  - Oversight will be provided to the Vendor Contact Program via periodic QA Audits and Surveillances of the vendors. This will ensure that: 1) vendors participating in our program are maintaining mailing and document distribution lists as requested, and 2) vendors not participating in our program are spot-checked for documents which have been transmitted.

  - Periodic effectiveness reviews of our Vendor Document Control Programs will be conducted.

  - Training needs will be evaluated and additional training programs implemented as required.

Summary

PSE&G was unable to access the three (3) Limitorque Maintenance Updates and ten (10) MOVATS Technical Notices which should have been retrievable through our Vendor Document Control System.

Upon assessment of this situation, root causes have been identified and immediate corrective actions taken.

To prevent recurrence and further strengthen our vendor information control process, PSE&G has committed to the actions identified above.

PSE&G recognizes that vendor information control is a very complex and extremely important issue which must be thoroughly and continuously addressed in order to operate our nuclear units safely and reliably.

To this end, PSE&G is committed to work with our vendors and the industry to demonstrate continued improvement in this area to our customers, regulators, and ourselves.

SUPPLEMENTARY INFORMATION - INSPECTION REPORT 354/91-80

1. NRC Comment: (Section 2.0, Para. 2, p. 4)

   - The scope of the program is limited to only active function valves. GL 89-10 and Supplement 1 recommended that all MOVs in safety-related systems be included in the MOV program scope.

Supplementary Data:

The scope of the program is not limited to only active function valves and encompasses a large population of valves based on the criteria outlined in Appendix A1 of Programmatic Standard NC.DE-PS.ZZ-0033 (Q). The basic criteria for including valves in the program is summarized as follows:

a) MOV is required to perform an active safety function;

b) Operability of the MOV is required by Technical Specifications; or,

c) MOV operation is required in the course of performing Operating, Abnormal and Emergency Operating Procedures.

We believe these criteria meet the intent of the Generic Letter recommendations.

2. NRC Comment: (Section 2.0, Para. 2, p. 4)

   - The licensee intends to use the BWR Owners Group recommendations for design basis reviews. However, the licensee has not performed a detailed review to determine the applicability of the owner's group recommendations to Hope Creek.

Supplementary Data:

The use of the applicable owners group methodologies, with regard to design basis flow and differential pressure calculations, is an inherently conservative approach which we believe does not require a detailed review. The program will consider any of the unique design attributes of the system and valve functions as part of the design basis determinations and will adjust the methodologies as appropriate.

3. NRC Comment: (Section 2.0, Para. 2, p. 4)

   - Inaccuracies of the diagnostic equipment have not been adequately addressed.

Supplementary Data:

At the time of the inspection, a draft Programmatic Standard was provided which described the MOV Program. The implementing Appendices will supplement this standard and will define total program implementation. It was always our intention to address the inaccuracies of diagnostic equipment through total program development. PSE&G has taken initial positive actions in this area. A conference was held at Artificial Island on August 30, 1991 to begin the process of validating the accuracy of the diagnostic equipment in use at Artificial Island. Attendees included representatives from the equipment manufacturer and also from two other regional utilities which utilize similar equipment.

The purpose was to pool efforts and resources with the regional utility users and the manufacturer to provide an efficient and cost effective accuracy determination program. In addition to this regional effort, PSE&G is also participating in the diagnostic equipment manufacturer's conference on this topic scheduled for the 9th, 10th and 11th of September. This topic is being actively addressed by PSE&G due to the sensitivity of the isuues.

4. NRC Comment: (Section 2.0, Para. 2, p. 4)

- Testing where "practicable" has not been clearly defined to preclude deviations from the intent of the generic letter.

Supplementary Data:

The MOV Programmatic Standard uses the term practical instead of practicable. The evolution of the MOV program will more clearly define the use of the term "practical" and its impact on the total population that can be safely tested at pressures at or near the maximum capability.

5. NRC Comment: (Section 2.0, Para. 2, p. 4)

- Periodic Verification has not been adequately addressed.

Supplementary Data:

Periodic re-verification of switch setting adequacy will be performed on a frequency not to exceed 5 years or 3 refueling outages as outlined in Section 4.4.4 of the MOV Programmatic Standard. This frequency will be evaluated and may be increase or decreased based upon MOV specific evaluations as data is compiled. The requirements for periodic setting verification following maintenance activities will also be clearly delineate

6. NRC Comment (Section 2.0, Para. 2, p. 4)

- The program does not address section h. of the generic letter relating to "failures", "corrective actions" and "trending".

Supplementary Data:

Section 4.6.1 of the MOV Programmatic Standard addresses Section h. of the Generic Letter.

7. NRC Comment: (Section 2.0, Para. 2, p. 4)

- The program does not contain sufficient details to demonstrate how the recommended schedules of the generic letter will be implemented.

Supplementary Data:

A schedule will be developed once the MOV population has been selected and prioritized as outlined in Appendices A1 and A2 of the MOV Programmatic Standard.

8. NRC Comment (Para. 3.1, p. 4)

There was a lack of communication between the Maintenance and Engineering departments. For example, the maintenance department adjusted the minimum required MOV thrust values provided by engineering by +10% to +20%. Neither maintenance nor engineering knew the source or exact reason for this adjustment. Also, neither organization could explain how diagnostic inaccuracies are addressed in the licensee's MOV program.

Supplementary Data:

An explanation of 10-20% target thrust window over the design minimum was provided to the inspectors during the inspection. This explanation was provided by the Engineering Department. The Maintenance Supervisor had properly deferred the question to the Engineering Department as being outside his area of expertise.

We take exception to the conclusion and its basis.

9. NRC Comment: (Para. 3.3, p. 5)

...Furthermore, the licensee had not evaluated or incorporated MOVATS Engineering Report 5.0 "Equipment accuracy summary" which provides recommendations on how to account for "rate of loading" effects.

Supplementary Data:

According to MOVATS personnel, Engineering Report 5.0 had never been disseminated to utilities.

The inspector was uncertain how this document had come into the Region's possession.

"Rate of loading" will be addressed in our program as indicated in Supplement 1, i.e., as a consideration in the use of test results for similar valves.

PSE&G is aware of and is actively involved with the industry in the resolution of instrument inaccuracy issues.

10. NRC Comment: (Section 3.5, Para. 2, p. 6)

Procedure NC.NA-AP.ZZ-0050 (Q), "Station Testing Program" does not clearly define the required post maintenance testing (PMT) following maintenance activities. Also, PMT for actuator replacement and valve packing adjustments did not require differential pressure or diagnostic testing.

Supplementary Data:

Attachment 4 (Page 4 of 6) of the subject procedure indicates the appropriate Post Maintenance Testing and Operability retest requirements for actuator replacement and valve packing adjustments. Possibly the inspectors did not have all pages of the procedure in question.

However, it is important to note that NC.NA-AP.ZZ-0050(Q) is an administrative procedure that is used in the Planning and Implementation process for guidelines. It is not a stand-alone document. In conjunction with the IST program and the Plant Technical Specifications, the System Engineer, Maintenance supervisor, Planner, and Licensed Operator, discuss and establish the retest requirements on a case by case basis.

Currently, based on our committed response to NRC Bulletin 85-03, and the interpretation of ASME Section XI requirements for Pre and Post Maintenance LLRT's, we do not necessarily perform diagnostic testing after a valve packing adjustment. This statement considers the requirement to maintain packing gland torque within the guidelines of our Chesterton Repack Program.

Based on the anticipated results of the program associated with generic letter 89-10, we do foresee a change to the requirements for differential pressure or diagnostic testing of motor operated valves.

10. NRC Comment: (Section 3.5, Para. 3, p.3) regarding the schedule for MOV overhaul.

Supplementary Data:

In conjunction with the existing trending program to assess historical motor operated valve failures, recommendation h. of Generic Letter 89-10 requires that an adequate trending program be established to analyze MOV failures and diagnostic test results to verify the adequacy of our Maintenance Program. We committed to the inspectors that recommendation h. of GL 89-10 would be adequately addressed in our final approved version of the programmatic standard for motor operated valve program.

In the draft project plan for the GL 89-10 MOV Program, the responsibilities for each member of the project team are delineated. One of the responsibilities listed for the Reliability and Assessment Department is to "evaluate the long-term maintenance requirements (rebuild) based on reliability centered maintenance (RCM) approach."

It is anticipated that the reliability assessment performed for MOVs will include an evaluation of vendor maintenance requirements, EQ program requirements, available condition monitoring and predictive maintenance techniques, and an interpretation of current industry practices. This approach is expected to satisfy the inspectors' concern of an adequate overhaul program/schedule.

11. NRC Comment: (Section 3.5, Para.4, p. 6)

The inspectors noted a concern with the general valve maintenance procedure HC.MD-GP.ZZ-002(Q) general inspection criteria (paragraph 5.2). They felt that this paragraph and accompanying check list/data sheet did not provided adequate guidance to perform meaningful inspection.

Supplementary Data:

PSE&G's position regarding this procedure as discussed with the inspectors was:

1. This is a "general" valve maintenance procedure.

2. The intent of this paragraph is to document a "general" overall visual inspection of a valve and operator for obvious signs of degradation, etc.

   Examples of degradation that would be typically looked for are: ~~loose~~ bolting, external leakage, etc.

3. This paragraph relies on training that a typical boiler repair mechanic would have to allow him to be qualified to repair valves.

The use of the words "general inspection criteria" in paragraph 5.2 was not meant to imply inspection to rigorous and absolute standards, rather the sense of a "general inspection check". We intend to add detailed inspection criteria consistent with the 89-10 program development.

12. NRC Comment: (Section 3.5, Para. 4, p.6)

The inspectors related a concern with the preventive maintenance procedure ([illegible]-PM.ZZ-004) requirement to maintain the lubricant in the limit switch gear box at 90 - 100% full.

Supplementary Data:

This 90% minimum requirement had been put into the procedure based on a letter from Limitorque Corporation to Bechtel, Inc. on April 11, 1984 in response to the specific concern about grease level.

When the inspectors provided us with a copy of an LER from a PWR utility, regarding a failure of an MOV due to excessive lubricant, we took immediate action. We contacted the Maintenance Department of the affected utility, and other utilities to discuss how to implement the practice of not filling 100%. We have already initiated a procedure revision to incorporate the lessons learned.

13. NRC Comment: (Section 5.0, Para. 2, p.9)

The inspectors noted that the vendor drawing for RHR valves HV-F015A and B indicated a direction of flow which appeared backwards for a globe valve.

Supplementary Data:

A walkdown of valve 1BCHV-F015A performed on 7/20/91 verified that the installed position is such that flow is under the seat. Anchor Darling was contacted on 7/22/91 and this information (i.e. installed position) was discussed and confirmed. A review of P&ID's Fabrication Isometrics and Installation Specifications have identified that these valves were installed with flow under the seat and the Anchor Darling valve drawing was not revised to reflect this installed condition. Based on the Bechtel installation document system, this was allowed since the highest tier document, in this case installation specification 10855-P205[illegible], specified how these valves were to be installed. We have initiated a "document only" design change, design change No. 4HC-0[illegible]5, to revise the valve drawing for 1BCHV-F015A&B to identify that flow is under the seat.

We have also reviewed all "safety related" globe valves used for throttling at Hope Creek for any similar drawing discrepancies (based on a computer run of all safety related globe valves from MMIS and reviewed P&ID's, Fabrication Isometrics, Valve Manufacturer drawings, etc.).

Eleven (11) other Valve Manufacturer drawings do not contain a "flow" arrow that explicitly identifies if flow is over or under the seat. We have verified that these valves are installed properly and we plan on revising these Valve Manufacturer drawings as part of design change No. 4HC-02345.

14. NRC Comment: (Section 6.0, Para. 4, p. 9)

"The licensee acknowledged that these issues will be properly addressed in their revised response to Supplement 3 of the generic letter as well as in the development and implementation of the GL 89-10 MOV Program."

Supplementary Data:

Our revised response to Supplement 3 will clarify our basis for concluding that the valves will function under design basis conditions and will address the questions contained in the June 25, 1991 Request for Additional Information.

ATTACHMENT 4

HOPE CREEK POST ACCIDENT SAMPLING SYSTEM (PASS) SEQUENCE OF EVENTS

Date(s) - 1991	Event
January-February	Third Refueling Outage.
Early March	Post outage testing - PASS operable.
March 15	Emergency drill - PASS had no detectable sample flow noted during system operation.
May 16	Training personnel noted low sample flow during PASS operations.
May 17	Chemistry received written emergency drill observation that PASS had no detectable sample flow on March 15. Chemistry supervision checked PASS operation and concluded that it was operable.
May 16-July 9	Training continued on PASS - low sample flow was still observed. Training informally communicated these findings to chemistry supervision.
July 12	Chemistry received training feedback form documenting PASS flow problems. Work Order (WO910712101) written.
July 19	Incident Report (91-111) initiated.
July 12-August 5	PASS inoperable during troubleshooting on system. Licensee determined that pipe sealant material caused the blockage.
August 5	PASS repaired and declared operable.