U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report Nos. 50-272/93-08
50-311/93-08
50-354/93-06

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: March 14, 1993 - April 17, 1993

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Inspection Summary:

This inspection report documents inspections of activities affecting public health and safety during day and backshift hours, including: operations, radiological controls, maintenance and surveillance testing, emergency preparedness, security, engineering/technical support, and safety assessment/quality verification. The Executive Summary delineates the overall inspection findings and conclusions.

EXECUTIVE SUMMARY

Salem Inspection Reports 50-272/93-08; 50-311/93-08

Hope Creek Inspection Report 50-354/93-06

March 14, 1993 - April 17, 1993

OPERATIONS (Modules 60710, 71707, 71710, 93702)

Salem: The licensee operated the Salem units safely. The inspector found the licensee's actions taken in response to the March 16, 1993, Unit 2 reactor trip to be appropriate and effective, as were their corrective actions and event follow-up. The Unit 2 seventh refueling outage was initiated following the March 16 reactor trip, and the inspectors determined the outage activities performed during the inspection period to be well planned, coordinated, and executed. Unit 1 operators were forced to reduce unit power level several times during the period due to marsh grass accumulation in the circulating water system. The inspectors observed good operator performance during these events and noted that Operations management conservatively managed unit power as a result of the environmental conditions.

Hope Creek: The licensee operated the Hope Creek unit safely. There were no significant challenges to plant operation.

Common: PSE&G conducted a fire protection drill during the inspection that involved assistance from offsite emergency response forces. The drill was a good exercise of the site Fire Protection and Security Departments, both of which responded well.

RADIOLOGICAL CONTROLS (Modules 71707, 93702)

Salem: Periodic inspector observation of station workers and Radiation Protection personnel noted good implementation of radiological controls and protection program requirements. The inspectors noted good performance in this area especially with respect to the Unit 2 refueling outage and its associated containment activities.

Hope Creek: Periodic inspector observation of station workers and Radiation Protection personnel noted good implementation of radiological controls and protection program requirements. An incident involving a non-licensed radioactive waste operator and apparent violation of licensee procedures is unresolved.

MAINTENANCE/SURVEILLANCE (Modules 61726, 62703)

Salem: Inspection in this area found good performance in the routine maintenance and surveillance activities performed at both Salem units. The licensee declared an Unusual Event at Unit 2 when a maintenance activity involving service water system piping replacement resulted in the inadvertent discharge of carbon dioxide gas in a vital area. The inspector concluded that the licensee properly responded to the event, but an unresolved item was opened pending the licensee's evaluation of the potential generic effects of the event at both Salem units. The inspector closed an open item after determining that PSE&G has acceptable programs to assure the control of expendable and consumable items.

Hope Creek: Two Technical Specification surveillance intervals were missed relating to the high pressure coolant injection system isolation function, and the main steam isolation valve sealing system valve stroke times. The latter issue is unresolved. The reactor recirculation pump end-of-cycle and anticipated transient without scram trip breakers were found operable and related surveillances were acceptable.

EMERGENCY PREPAREDNESS (Modules 71707, 93702)

The inspectors observed and participated in (1) portions of an unannounced off-hours emergency preparedness drill at Hope Creek that the licensee conducted to especially test their automated callout system and (2) in a routine monthly drill conducted at Salem. The inspectors determined both drills to be well conducted and an effective exercise of the licensee's Emergency Plan.

SECURITY (Modules 71707, 93702)

The inspectors determined that the licensee appropriately implemented security program requirements. The inspector concluded that the licensee demonstrated a proactive approach relative to severe winter storm planning and appropriately compensated for any degraded conditions. The inspector also noted that the PSE&G Security Department performed well in the April 14, 1993, fire protection drill that required the security force personnel to process offsite emergency response forces into the Artificial Island protected area under simulated emergency conditions.

ENGINEERING/TECHNICAL SUPPORT (Modules 37828, 71707)

Salem: The inspectors noted that engineering personnel properly prioritized work activities. The licensee engineering staff provided a good evaluation and safety-conscious resolution when the Salem units' control air system outboard containment isolation valves were determined to be outside their design basis. The licensee identified that the emergency diesel generator cooling water flow control valves had been installed with improper setpoints which constituted a condition outside their design basis. The inspector noted good engineering response to the discovery by the licensee, although the item remains unresolved pending the licensee's evaluation of the past effect the setpoint error could have had on the generators' operability under design conditions.

Executive Summary

Hope Creek: The inspectors noted that engineering personnel properly prioritized work activities. A violation regarding the lack of testing of the standby start feature on Reactor Building filtration, recirculation and ventilation system fans was closed.

SAFETY ASSESSMENT/QUALITY VERIFICATION (Modules 30702, 40500, 71707, 90712, 90713, 92700, 92701)

Salem: The inspectors found generally positive acceptance by Salem operators of the re-unification of the Salem operating crews and concluded the new shift schedule should have a positive effect on reactor operator and equipment operator morale. The inspectors closed a previously open item when they concluded that the containment isolation function for the feedwater system continues to be in accordance with Technical Specification requirements.

Hope Creek: Licensee follow-up to plant events was thorough and effective.

Common: The NRR project managers (PMs) for Salem and Hope Creek inspected the licensee's 10 CFR 50.59 program. The PMs found several discrepancies in a relatively small sample size, which indicated a weakness in the program, and an apparent violation of 10CFR50.59.

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DETAILS

1. SUMMARY OF OPERATIONS

1.1 Salem Units 1 and 2

Unit 1 began the inspection period operating at 100% power. On March 15, 16 and 23, 1993, and through April until the end of the inspection period, the unit operated at mostly reduced power levels due to the effects of seasonal marsh grass accumulation on the circulating water system and the main condensers (See Section 2.2.1.C).

Unit 2 also began the period at full power but tripped on March 16, 1993, due to a steam generator feed pump trip (See Section 2.2.1.A). PSE&G elected to maintain the unit shutdown and enter the unit's seventh refueling outage slightly ahead of schedule. The plant reached Mode 5 (Cold Shutdown) on March 15, Mode 6 (Refueling) on March 25, and the unit was de-fueled by April 5 and remained so through the end of the inspection period.

1.2 Hope Creek

The Hope Creek unit operated at power during the period.

2. OPERATIONS

2.1 Inspection Activities

The inspectors verified that Public Service Electric and Gas (PSE&G) operated the facilities safely and in conformance with regulatory requirements. The inspectors evaluated PSE&G's management control 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. The inspectors performed normal and back-shift inspections, including deep back-shift (39 hours) inspections.

2.2 Inspection Findings and Significant Plant Events

2.2.1 Salem

A. Salem Unit 2 Reactor Trip

On March 16, 1993, at 11:06 a.m., the Salem Unit 2 reactor automatically tripped from 100% power. At 11:04 a.m. the No. 22 steam generator feed pump (SGFP) had tripped on low suction pressure, and the operators initiated a turbine generator runback, in an attempt to reduce power to 60%. However, before the runback was completed the reactor tripped on No. 24 steam generator (SG) low-low level. The licensee informed the resident inspector, and the inspector arrived in the control room approximately five minutes after the reactor trip. The licensee subsequently reported the event to the NRC Operations Center.

Prior to the No. 22 SGFP trip, reactor and SG pressures and temperatures were stable. Control room operators received No. 22 SGFP "High-vibration" alarm and noticed that No. 22 SGFP had tripped. Operators initiated a turbine load reduction. All SG levels trended downward. An automatic reactor trip occurred when No. 24 SG reached its low-low level trip setpoint.

The licensee entered the reactor trip procedures, Emergency Operating Procedure (EOP) - Trip-1 and 2, which required initiation of a manual steamline isolation because a high auxiliary feedwater (AFW) flow rate resulted in lowering primary system average temperature. Systems responded normally to the trip with the following exceptions: (1) the No. 24 SG feed regulating valve (24BF19) failed open, and (2) the No. 23 AFW pump restarted even though no valid start signal was present. The licensee cooled down the plant and entered Mode 5 (Cold Shutdown). Licensee management elected to commence the Unit 2 seventh refueling outage four days ahead of its scheduled start date. The licensee formed a Significant Event Response Team (SERT) to determine causes and corrective actions for the reactor trip.

The licensee's investigation determined the proximate cause of the event to be the failure of a condensate polishing (CP) system pressure control switch due to water intrusion from a leaking valve. This caused the 24CP2 valve to open, diverting SGFP suction flow and resulting in the actuation of the low suction pressure switch on the No. 22 SGFP. The licensee confirmed the cause during a condensate system test on March 18, 1993, and replaced the failed pressure control switch. The licensee determined the root cause to be a management/QA deficiency for failure to take timely actions to correct the leaking condensate polishing valve. The licensee determined that the 24BF19 valve was held in the open position by a piece of metal pipe, which was apparently from a broken chemical feed line upstream of the valve. This pipe failure was due to an original construction deficiency. A check of Unit 1 did not note the same deficiency. Licensee investigation into the 23 AFW pump unexpected restart determined that a start/stop valve solenoid failure allowed the steam admission valve (MS132) to stay open. Thus, the pump restarted without an actual start signal. The licensee replaced a faulty auto start relay contact. The licensee submitted Licensee Event Report (LER) 93-05 for this event.

The inspector reviewed the operations logs and control room recorders, verified EOP implementation, interviewed onshift operators, and reviewed and discussed the event with the SERT team and plant management. The inspector reviewed the SERT report, LER 93-05 and the AD-16 procedure (post reactor trip review). The inspector found the licensee's actions taken in response to the event appropriate and effective. The inspector's evaluation of the licensee's corrective actions and event follow-up determined them to be appropriate.

B. Unit 2 Refueling Activities

On March 29, 1993, the licensee commenced core offload for the Unit 2 seventh refueling outage. The inspector observed fuel handling activities from the fuel handling building, containment refueling platform and the control room. The inspector noted good coordination, communication and cooperation between the licensee and the Westinghouse fuel handlers. The defueling process was performed with precision and professionalism. The inspector noted that the fuel handling supervisors emphasized attention to detail, quality control, and good radiation work practices, even at the expense of expediency. The inspector interviewed the refueling senior reactor operator, the equipment operator, the upender operators, and the Westinghouse shift engineer. All personnel were cognizant of their duties and responsibilities and very knowledgeable of the procedure and process.

The licensee temporarily stopped refueling activities on two occasions. At 6:02 a.m. on March 31, 1993, the spent fuel pit (SFP) fuel transfer system upender operator inadvertently lowered the upender while fuel assembly U-47 was being withdrawn. The fuel assembly was raised approximately 75% out of the upender frame when the upender operator accidentally bumped the "frame down" button on the fuel transfer system control console. The upender operator immediately pushed the "frame stop" button. The upender frame moved approximately five inches down. The spent fuel tool operator, stationed on the SFP bridge, notified the refueling shift supervisor in containment. The shift supervision assessed the situation, then directed the frame upended, and the fuel assembly removed and placed in the spent fuel pit. The spent fuel tool operator performed a visual inspection of the fuel assembly and upender basket. No damage was detected. The licensee resumed fuel handling at 6:14 a.m. on March 31. On April 6, the U-47 fuel assembly was the first assembly to undergo an ultrasonic test and visual test. The licensee and Westinghouse fuel representatives performed a thorough examination of the assembly and concluded that no damage was done to the assembly.

At 4:30 p.m. on March 31, 1993, fuel handling activities were halted when a weld broke on the fuel transfer system conveyor car roller chain. The conveyor car supports the fuel assembly as it is transported horizontally through a tracked tubular passageway connecting the reactor side lower cavity and the fuel handling building spent fuel pit. The track-mounted conveyor is driven by a sprocket and chain drive mounted between the tracks. The roller chain is welded to the bottom of the conveyor car and engages the floating drive sprockets on the reactor side. The broken weld allowed the roller chain to disengage from the sprocket, preventing the transfer of fuel assemblies between the reactor cavity and the SFP. The licensee evaluated the situation and decided the chain could best be repaired by removing the conveyor car from the lower cavity and rewelding the roller chain on the containment refueling deck. The licensee removed, repaired and replaced the trailing 20-foot portion of the conveyor car. During this repair the licensee discovered five broken welds on the leading 15-foot portion of the conveyor car. These welds were also repaired. The licensee attributed the broken weld to roller chain wear and misalignment over time. The core offload resumed at 6:44 a.m. on April 5, and was completed later that morning at 8:45 a.m.

The inspector discussed the two events, the fuel transfer system design and operation, and the licensee's work plan with the cognizant reactor engineer. The inspector concluded that the two events were unrelated, the work activities were well-planned and the licensee exercised appropriate safety radiological precautions. The inspector discussed the inadvertent upender operation incident investigation with the licensee. The inspector noted that the resultant consequences were minimal and the licensee's immediate actions were acceptable.

Overall, the inspector found the Unit 2 offload appropriately planned, coordinated, executed, and documented.

C. Unit 1 Power Reductions Due to Circulating Water Concerns

At various times during and through the majority of the second half of this inspection period, PSE&G operated Salem Unit 1 at a reduced power level due to concerns related to the circulating water system. The circulating water system uses the Delaware River as a source of water to condense the steam exiting the main turbine. For environmental reasons, the State of New Jersey limits the allowed 24-hour average temperature rise for circulating water to a maximum of 27.5°F. At various times during the inspection period, because of silt and grass collecting in condenser waterboxes, Salem Operations isolated individual waterboxes to clean them and lower the temperature rise across all of the waterboxes. In order to avoid approaching the temperature limits, Salem operators reduced unit power while waterboxes were isolated. During the second half of the inspection period, circulating water problems were aggravated by a large amount of marsh grass collecting in the river by the circulating water intake structure. This marsh grass phenomenon occurs this time of year at Artificial Island and was worsened this year by several rain storms. As a result of grass accumulation on the circulating water traveling screens, circulating water flow was decreased and, at times, circulating water pumps tripped off due to low suction pressure.

The effects of the circulating water system events presented a challenge to the Salem operating crews as they were forced to adjust plant power to match the available circulating water system configuration and to adhere to the water temperature-rise limits. The operators reduced power to approximately 90% on March 15, 16 and 23, 1993, to accommodate circulating water system restrictions and, during the first week of April, power level was varied several times between 60% and full power. From that time through the end of the report period, Operations management made the decision to maintain power level between 70% and 80% until the environmental conditions cleared.

The resident inspector observed the challenge these conditions presented to the Salem operators, especially on the night of April 14, when up to four of the six circulating water pumps were lost from service. The inspector noted good performance on the part of the operating crews, and the assistance they received from Salem maintenance and site services personnel, in operating the unit and managing to keep it on line. The inspector also noted Salem Operations management's decision to operate at a lower power level during these conditions to be conservative and prudent.

2.2.2 Hope Creek

The Hope Creek unit remained at or near full power during the period. The inspectors monitored steady-state unit operations and performed routine inspection activities. The inspectors concluded that the licensee safely operated and maintained the unit during this inspection period.

2.2.3 Common

A. Fire Protection and Security Offsite Assistance Drill

On April 14, 1993, PSE&G conducted an after-hours drill involving the site Fire protection and Security Departments and which was designed to require assistance from offsite emergency response forces. The drill scenario consisted of a simulated fire and personnel injuries at the new warehouse facility inside the protected area at Artificial Island. The primary focus of the drill was to exercise the PSE&G Nuclear Fire Protection Department and its ability to combat the fire, locate and assist the injured personnel, and to solicit the help of and integrate the participation of offsite forces. The drill also required the participation of the PSE&G Site Protection Department in that site security forces were required to process the offsite response personnel into the protected area via emergency procedures.

On the day of the drill, the PSE&G Nuclear Security Manager and the Senior Nuclear Fire Protection Supervisor briefed the NRC resident inspectors on the drill scenario and expectations for licensee performance. Following the performance of the drill, the inspector discussed the drill with the Senior Fire Protection Supervisor, reviewed the licensee's drill critique and viewed a video tape of various aspects and highlights of the drill. In addition to reviewing the drill results, the inspector toured the site Fire Protection Department's facilities and equipment and discussed the Department's capabilities with the Supervisor. The inspector concluded that the drill had been a worthwhile exercise of the PSE&G Fire Protection and Security Departments and that the Fire Protection Department maintains an ability to respond to site emergencies very well.

3. RADIOLOGICAL CONTROLS

3.1 Inspection Activities

The inspector verified on a periodic basis PSE&G's conformance with the radiological protection program.

3.2 Inspection Findings

3.2.1 Salem

A. Containment Tours

The inspector periodically toured the Salem Unit 2 containment during the current refueling outage period. Items checked included access controls, use of anti-contamination clothing, worker radiation practices, dosimetry and exposure controls, decontamination procedures, tool control, and work in progress. The inspector found the radiation protection personnel very knowledgeable, extremely visible in and out of containment, actively involved in radiological controls, and keenly interested in minimizing exposure to workers. The inspector observed the licensee's use of appropriate radiological precautions and controls during the core offload and weld repair of the refueling conveyor car. The inspector noted good radiation work practices and a good ALARA consciousness.

3.2.2 Hope Creek

A. Improper Personnel Entry Into Radiological Controlled Area (RCA)

On Monday, April 12, 1993, during an informal audit of radioactive waste operators' time sheets and RCA access records, a licensee supervisor noted an apparent discrepancy between one operator's stated work start time and RCA access times. When questioned about the discrepancy, the operator stated that he had reached his assigned work area, the radwaste control room, after transiting portions of the turbine building, including the maintenance "hot shop", which is in the RCA. About one hour later, he then reported to the RCA access control point to obtain his dosimetry. The licensee determined that this method of entry had occurred twice, once on Saturday, April 10, and again on April 12.

The licensee concluded that these acts were violations of administrative procedures NC.NA.AP-ZZ-024,"Radiation Protection Program," and HC.SA.AP-ZZ-0046,"Radiological Access Program." The individual's dosimetry was immediately pulled and access to the RCA barred. Through a review of applicable access records and time sheets of a number of other radwaste workers, the licensee determined that this violation of RCA requirements was apparently an isolated case, as no other such discrepancies were found. Longer term actions were being developed when the inspection period ended.

The inspector discussed this incident with health physics and plant management. The inspector was concerned that these violations of station procedures had apparently been committed by an experienced radiation worker on at least two occasions. This issue is unresolved pending completion of the licensee's corrective actions and NRC review **(URI 50-354/93-06-01).**

4. MAINTENANCE/SURVEILLANCE TESTING

4.1 Maintenance Inspection Activity

The inspectors observed selected maintenance activities on safety-related equipment to ascertain that the licensee conducted these activities in accordance with approved procedures, Technical Specifications, and appropriate industrial codes and standards.

The inspector observed portions of the following activities:

Unit	Work Order(WO) or Design Change Package (DCP)	Description
Salem 1	Various	Circulating water and main condenser repair and cleaning
Salem 2	WO 9303311149	Repair of fuel transfer system conveyor car
Hope Creek	WO 930301168	Replace transmission seal on Technical Support Center (TSC) chiller BK403

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.

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

Unit	Procedure No.	Test
Salem	S2.OP-ST.DG-0002(Q)	Unit 2 Emergency Diesel Generator 2C Operability Test
Hope Creek	OP-ST.KJ-001	"A" Emergency Diesel Generator Monthly Surveillance Test
Hope Creek	OP-IS.BD-001	Reactor Core Isolation Cooling (RCIC) Jockey Pump 92-Day Inservice Test

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. Unusual Event at Unit 2 Due to Toxic Gas Discharge

On April 3, 1993, with Unit 2 in the Refueling Mode, Salem Maintenance workers were performing service water piping replacement work in the 78-foot elevation mechanical penetration area. Due to silt clogging of the pipe, and unknown by the workers, the piping to be replaced could not be properly drained. When the workers cut the line, water sprayed into the overhead of the mechanical penetration area. Approximately five hours later, the carbon dioxide fire protection system discharged into the adjacent 78-foot elevation electrical penetration area. The licensee immediately evacuated the effected area and declared an Unusual Event due to the discharge of a toxic gas which was a threat to personnel in a vital area. The Salem operating crew properly notified the NRC Operations Center and resident inspector of the event and the emergency declaration. The licensee terminated the Unusual Event approximately one hour after the event, upon restoration of the electrical penetration area to normal habitability conditions and determining no personnel injuries had resulted from the event.

Following a subsequent investigation, the licensee determined the cause of the inadvertent discharge of the carbon dioxide system to be water intrusion into the carbon dioxide system control panel located in the mechanical penetration area. The licensee's investigation also revealed that the only potentially significant plant equipment effected by the event was the "A" reactor vessel level indication channel, which was not required to be in service at the time of the event but was adversely affected by the cold temperature resulting from the discharge. The resident inspector verified that PSE&G had properly implemented their Emergency Plan and that the proper personnel protection actions had been taken subsequent to the event. The inspector also discussed the event with the Salem Quality Assurance inspector who led the licensee's investigation, and concluded that PSE&G had properly responded to the event and determined its root cause in a proper fashion. At the end of the report period, however, the licensee had yet to determine if the ability of the water to intrude and collect in the affected control panel was a design flaw or a result of improper construction or installation. Until this question has been resolved and corrective measures identified, this item will remain open **(URI 50-272 and 311/93-08-01).**

B. Open Item Follow-up

(Closed) Unresolved Item 50-272&311/91-16-01: Review of existing programs to assure acceptable control of safety-related expendable and consumable items. The inspector reviewed the licensee's procedure that certifies reactor and secondary plant bulk chemicals. The procedure, No. SC.CH-CA.ZZ-0401(Q), provides sampling and analysis requirements for selected bulk chemicals that may be a source of impurities for plant systems. The inspector also determined that expendable and consumable items (chemicals) are classified, labelled and controlled per the requirements specified in Nuclear Administrative Procedure (NAP) No. 38, "Chemical Control Program." The inspector concluded that the licensee has acceptable programs to assure control of expendable and consumable items, and therefore closed this unresolved item.

4.3.2 Hope Creek

A. Missed Surveillances on Main Steam Isolation Valve Sealing System (MSIVSS) Valves and High Pressure Coolant Injection (HPCI) System

On March 23, 1993, during a review of several outage work orders for which the original retest activities were not available, the licensee discovered that several valves in the MSIVSS had not been surveilled as required by Technical Specification (TS) 4.0.2. Both valves (HV-2512B and HV-5829B) were refurbished during the fourth refueling outage (September-November 1992). At that time, the post-maintenance tests apparently had not required timing the valve stroke. The licensee could find no documentation that the surveillance procedure (OP-IS.KP-103) had been performed until March 1993. The licensee concluded that both valves were operable as, at the time of discovery, both were within their current surveillance frequency.

LER 93-01 discussed the circumstances surrounding a missed Technical Specification required surveillance on the high pressure coolant injection system (HPCI) isolation delta temperature instrumentation. Due to a procedure inadequacy covering the use of primary and backup instruments, technicians failed to perform a channel calibration on the A2 logic channel after the primary instrument had been repaired during the fourth refueling outage. The inspectors noted that the safety significance of this event was minimal as other tests performed before, during and after the surveillance was missed indicated that the A2 channel was capable of performing its isolation function. Related HPCI isolation logic channels were also functional. The licensee's corrective actions included performing the appropriate logic channel surveillance and procedural revisions to identify requirements for spare channel usage and restoration to normal configuration. The inspector noted that this LER was generally well-written. However, in noting that there had been one similar occurrence, the licensee incorrectly referenced LER 86-09 (The correct LER number was 89-06.). The inspector brought this minor discrepancy to the licensee's attention, who indicated that a corrected LER would be submitted. The LER remains open.

The inspector reviewed the event and concluded that there was minimal safety significance to the missed surveillances. However, the inspector noted that this was a second instance of a missed TS required surveillance during this reporting period. The licensee's review of these events is ongoing. These events are unresolved pending completion of the licensee's review and implementation of corrective actions **(URI 50-354/93-06-02)**.

B. Recirculation Pump Trip Logic Surveillance

On March 4, 1993, the licensee at the Washington Nuclear Power Station, Unit 2 (Hanford) reported (reference EN No. 25190) to the NRC that the end-of-cycle recirculation pump trip breakers were inoperable due to never having been surveilled. The licensee discovered this fact during a design review. Because of the similarities between Hope Creek and Hanford regarding the reactor recirculation systems, the inspector reviewed PSE&G's surveillance procedures and discussed this event with licensee operations and maintenance supervision. The inspector determined that both the anticipated transient without scram (ATWS) breakers and the end-of-cycle pump trip breakers (two breakers for each function) were tested to demonstrate operability. Each breaker is tested individually with the control logic circuits tested from each process input to the breaker trip coil. That the breaker trips open when its associated trip coil energizes is also demonstrated. Based on this review and discussion, the inspector concluded that Hope Creek's recirculation pump trip breakers were operable and that the surveillance procedures were adequate to demonstrate operability.

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, the inspector reviewed licensee event notifications and reporting requirements per 10CFR50.72 and 73.

5.2 Inspection Findings

A. Off-Hours Drill at Hope Creek

In order to evaluate the effectiveness of their emergency response organization's automated callout system and to demonstrate the ability to activate their emergency response facilities (ERFs) within an hour of notification of emergency responders, PSE&G conducted an unannounced off-hours emergency preparedness (EP) drill at Hope Creek early on the morning of March 25, 1993. As part of the drill, the licensee contacted the appropriate personnel via their pagers, and those employees reported to their assigned positions at an ERF (either the Hope Creek Operations Support Center, the Hope Creek Technical Support Center, the Emergency Offsite Facility or the Emergency News Center).

The NRC resident inspector was appropriately notified by the Hope Creek control room communicator during the drill and, subsequent to the drill, discussed the drill conduct and results with the PSE&G EP Manager and his staff. The inspector observed portions of the licensee's automated callout system, reviewed the accountability logs kept by PSE&G during the drill, and determined the drill had been appropriately conducted and that PSE&G had accomplished the requirements of their Emergency Plan for ERF manning.

B. Routine Emergency Preparedness Drill Conducted at Salem

On March 31, 1993, the PSE&G Emergency Preparedness (EP) organization conducted a routine monthly drill at the Salem facility. The drill involved the simulated sabotage of the Salem service water intake structure and a subsequent loss-of-coolant accident, which required the licensee to man the Salem Operations Support Center, the Salem Technical Support Center (TSC) and the Emergency Offsite Facility, and to declare a General Emergency per their Event Classification Guide.

The NRC resident staff participated in the drill at the TSC and the Salem simulator control room and determined that PSE&G personnel followed the appropriate procedures and performed well during the drill, and that the drill provided a good exercise of the licensee's Emergency Plan and organization.

6. SECURITY

6.1 Inspection Activity

PSE&G's verified regularly the conformance with the security program, including the adequacy of staffing, entry control, alarm stations, and physical boundaries.

6.2 Inspection Findings

A. Security Operations During the March 12-15, 1993 Winter Storm

The inspector reviewed the licensee's security operations during the severe winter storm that occurred March 12-15, 1993. (The inspector previously reviewed storm preparations and associated conduct of plant operations in NRC Inspection 50-272, 311, 354/93-02.)

The licensee initiated plans for staffing and reviewed security plan contingencies prior to the storm arrival on March 12, 1993. During the storm, security staffing was maintained by augmenting the onshift personnel, including holding personnel beyond their shift change in order to sleep. During the storm, the licensee compensated for degradations that occurred to portions of the security hardware caused by high winds and rain/snow.

The inspector discussed these security operations with security management personnel and with selected guard force members. The inspector also reviewed a licensee security report regarding this storm event. The inspector concluded that the licensee demonstrated a proactive approach to storm planning and appropriately compensated for the degraded conditions.

B. PSE&G Offsite Assistance Drill

On April 14, 1993, PSE&G conducted an after-hours drill of the site Fire Protection and Security Departments which required the solicitation and integration of offsite emergency response forces (see Section 2.2.3.A of this report). The inspector concluded that the site security personnel had performed well in expediting the in-processing of the required offsite personnel per emergency procedures as part of the drill scenario.

7. ENGINEERING/TECHNICAL SUPPORT

7.1 Salem

A. Unit 1 and 2 Containment Isolation Valves Determined to be Outside Their Design Basis

During the review of a design change to replace the solenoids for the control air system air-operated outboard containment isolation valves on both Salem units, the licensee determined, on March 4, 1993, that the valves were not as described in the Salem Updated Final Safety Analysis Report (UFSAR). The UFSAR states that automatic isolation valve closures are fail safe, i.e. closure is initiated, upon loss of voltage and/or control air. The identified valves, which are normally open, fail "as-is" on loss of 125 VDC power to their solenoid actuators; the solenoids must be energized to open the valves and to close them. The valves are designed to perform their isolation function upon receipt of a Phase A isolation signal, but the valves would be unable to close if 125 VDC power was not available.

Upon discovery of this condition, which is an as-built condition for both units, the licensee initiated an engineering evaluation to determine if this as-built configuration is appropriate with no changes. Factors considered in the licensee's evaluation were: the reliability of the 125 VDC electrical system; the presence of the inboard control air containment isolation valves and the fact that they are mechanical check valves; the availability of the two Emergency Control Air Compressors to maintain pressure in the control air header and the fact that minimum header pressure would be 65 psig, greater than containment design pressure of 47 psig; and that the outboard isolation valves still go closed upon loss of air pressure. The licensee's evaluation concluded that the valves' fail as-is configuration provides a level of safety consistent with 10CFR50, Appendix A, General Design Criterion 56, which requires, in part, that "... upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety." Based on that conclusion, the licensee prepared a Justification for Continued Operation (JCO), performed a 10CFR50.59 Safety Evaluation, reviewed and approved both at a Station Operations Review Committee (SORC) meeting, and submitted a request for licensee amendment to the NRC, in accordance with 10CFR50.90, in order to change the description of these valves in the UFSAR.

When the licensee originally identified the above identified discrepancy, they properly notified the NRC Operations Center in accordance with 10CFR50.72 and the resident inspector. The resident inspector discussed the condition with PSE&G engineering staff, examined the licensee's JCO and 10CFR50.59 evaluations, attended the related SORC meeting and reviewed PSE&G's Licensee Event Report (See Section 9.1) and 10CFR50.90 submittals. The inspector concluded that while PSE&G's lack of awareness of the Salem plants' as-built configuration was a weakness, but also that the licensee performed well in evaluating and resolving this issue and that the as-built configuration of the control air outboard containment isolation valves did not adversely affect the safe operation of the Salem plants.

B. Emergency Diesel Generator Cooling Water Flow Outside Design Basis

During service water piping upgrade work on the Unit 2 Emergency Diesel Generators (EDGs) on April 7, 1993, the licensee found an error in the setpoint of the differential pressure controllers for the valves which modulate service water flow to the EDG jacket water coolers and lube oil coolers at both Salem units. The field setpoint matched the PSE&G system description, i.e. a 6 psig drop across the coolers. PSE&G believed this value to be the value specified by the manufacturer's design, however, the manufacturer had specified this value for each cooler, not for the total pressure drop across both coolers in series as was found. The licensee determined the result of this error to be an approximate 16% reduction in the 700 gallon per minute design flow rate of service water through the coolers, and this resulted in a conservative determination of the EDGs only being operable if service water temperatures remain below 60°F. Once this condition had been identified, PSE&G made the proper notifications to the NRC, generated design change requests to properly set the controller setpoints, and initiated an engineering evaluation to determine the historical design basis significance of the situation.

The NRC resident inspector discussed the discrepancy with PSE&G engineering and determined that the discrepancy did not immediately impact EDG operability, in that river water temperature did not rise above 60°F prior to the implementation of the flow controller design change. The inspector also verified through licensee data that the EDGs had not experienced any heat load problems during their lifetime and that the design changes were adequately implemented in a timely manner. By the end of the inspection period, the licensee had not completed the evaluation of the past effect the setpoint error could have had on EDG operability under design conditions. Until that evaluation has been reviewed by the NRC, this item will remain open **(URI 50-272 and 311/93-08-02).**

7.2 Hope Creek

A. Open Item Follow-up

(Closed) Violation (50-354/92-03-04); Inadequate Filtration, Recirculation and Ventilation system (FRVS) Testing. On July 17, 1992, the licensee responded to a Notice of Violation (NOV) involving FRVS surveillance testing in which the automatic start function of the standby ventilation unit was not periodically tested. The licensee committed to a number of corrective actions, as detailed in their response to the NOV (Letter NLR-N92097, dated July 17, 1992). The inspector reviewed the licensee's response and ensuing corrective actions and determined that:

- The licensee modified surveillance procedure HC.OP-ST.SM-002, "Primary Containment Isolation System/Reactor Building and Refuel Floor Containment Isolation Functional Test-18 Months," to include testing of the control logic for both the auto-lead and standby functions of the FRVS fans, including the proper operation of the two-minute time delay component. The procedure also referenced the appropriate acceptance criteria.

- The licensee performed an evaluation of plant systems to identify any system transfer function whose failure to transfer could result in a loss of the system's safety function. The report was comprehensive and thorough. The review identified one similar instance. The licensee properly documented their finding in Incident Report 92-184. Their corrective actions were appropriate. The licensee had also implemented a number of other recommendations effecting non-safety related systems.

- The licensee performed a test of the auto-lead and standby functions of the FRVS fans during the eighteen month surveillance tests on the A and B emergency diesel generators in October 1992. The response time for both standby fans was 120.5 seconds (115-125 seconds was required).

Based on the foregoing, the inspector concluded that the licensee had acceptably addressed the issues cited in the violation and therefore closed the violation.

8. SAFETY ASSESSMENT/QUALITY VERIFICATION

8.1 Salem

A. New Shift Schedule for Salem Operations Nuclear Control Operators and Equipment Operators

On April 4, 1993, the Salem Operations Department re-aligned the shift schedule of the reactor operators (ROs) and equipment operators (EOs) such that the ROs and EOs would be working a 12 hour shift schedule to match the rotation schedule of the shift senior reactor operators (SROs). The SROs had been placed on the 12 hour shift rotation on November 15, 1992, but the ROs/EOs maintained an 8 hour rotation due to their union's objection to the 12 hour schedule (see NRC Inspection 50-272 and 311/93-01). Since that time, enough Salem ROs/EOs were attracted to the 12 hour rotation that the union dropped its objection, and Salem management obliged the ROs/EOs and re-united the ROs/EOs and the SROs on common shift schedules.

The NRC resident staff noted that the split-shift configuration of the Salem operating crews had not adversely effected licensed operator individual or team performance, and the shift overlap had, in fact, helped to foster Operations Department unity. In discussions with licensee operations following the April 4 change, the inspector found generally positive acceptance of the shift re-unification and concluded the new schedule should have a positive effect on RO/EO morale. The inspector will continue to monitor the transition to the new shift schedule and its effect on operator performance.

B. Open Item Follow-up

(Closed) Unresolved Item 50-272&311/92-01-05: Concerns associated with environmental qualification (EQ) for the feedwater system stop check isolation valves (BF-22s). The inspector found that BF-22 valves do not currently meet all EQ requirements to function as containment isolation valves (CIVs). For the interim, the licensee will continue to rely on the main and bypass feedwater regulating valves (FRVs) to function as the CIVs (per Technical Specifications). The licensee plans to fully qualify the BF-22s and then revise the Technical Specifications to replace the FRVs with the BF-22s for CIV purposes. The inspector concluded that the containment isolation function for the feedwater system continues to be in compliance with Technical Specifications. This item is closed.

8.2 Hope Creek

Licensee follow-up to plant events was thorough and effective.

8.3 10 CFR 50.59 Program inspection

A. Overview and Objective

The NRR project managers (PMs) for both Salem and Hope Creek inspected the licensee's Safety Evaluation Program (10 CFR 50.59 program) from February 23 through 26, 1993, and March 30 and 31, 1993. The PMs performed the inspection in accordance with Inspection Procedure 37001, "10 CFR 50.59 Safety Evaluation Program," issued December 29, 1992.

The objective of the inspection was to verify that the licensee implemented a safety evaluation program in conformance with 10 CFR 50.59, "Changes, Tests and Experiments" (CTEs). The objective was accomplished by (1) reviewing the licensee's procedures to verify that they conform to the 10 CFR 50.59 rule; (2) reviewing the licensee's training program; and (3) reviewing a sample of the licensee's 10 CFR 50.59 reviews and safety evaluations (SEs). The PMs noted that the licensee performs applicability reviews to determine whether 10 CFR 50.59 applies to a proposed CTE, and performs a safety evaluation (when it is determined that 10 CFR 50.59 applies) to determine whether the proposed CTE involves an unreviewed safety question. Accordingly, the PMs reviewed a sample of CTEs that receive an applicability review and subsequently required a Safety Evaluation; and a sample of CTEs that were reviewed for 10 CFR 50.59 applicability but did not require a 10 CFR 50.59 Safety Evaluation, as determined by the licensee. A list of the CTEs that were reviewed by the PMs is contained in Attachment 1.

B. Procedure Review, Salem/Hope Creek

Administrative Procedure NC.NA-AP.ZZ-0059(Q), Revision 0, "10 CFR 50.59 Reviews and Safety Evaluations" (NAP-59), is the licensee's governing document for 10 CFR 50.59 reviews and safety evaluations. The procedure was previously reviewed as documented in inspection reports 50-272/91-26, 50-311/91-26 and 50-354/91-19.

During this inspection the PMs reviewed NAP-59 in accordance with the guidance provided in Inspection Procedure 37001. The PMs determined that NAP-59 is well written, comprehensive, and adequately addressed implementation of the 10 CFR 50.59 rule.

C. Training Review, Salem/Hope Creek

PMs determined that the training program is excellent, overall. The program provided an in-depth and extensive discussion of the 10 CFR 50.59 rule, NAP-59, and recent NRC and industry 10 CFR 50.59 guidance. However, the PMs detected an element where the guidance given by the training department is incorrect and may cause inadvertent violation of the 10 CFR 50.59 rule.

Training module 0905-002.14B-5059ZZ-00, "10 CFR 50.59 Training," contains an "open-reference" exam that cites an incorrect answer to a posed question. Specifically, question number 1.19 requires students to give examples of when they must clearly answer "YES" to the question, "Does the proposal change the facility as described in the SAR [i.e., safety analysis report]?" The provided answer states,

> "There are NO cases where, categorically, we must answer this question, 'yes.' Management stresses the fact that we must think smart, and look at everything we review from the perspective of 'can this change impact the safe operations of the plant?'"

The PMs determined that this answer is incorrect and misleading since it changes the scope of the question from the intent of 10 CFR 50.59. The PMs noted that the 10 CFR 50.59 rule is intended to be applied in two steps. First, determination of whether 10 CFR 50.59 applies [The 10 CFR 50.59 rule applies if the licensee is changing a structure, system, or component (SSC) or a procedure described in the licensee's final SAR (FSAR) and if the FSAR description of the SSC (or procedure) being changed would be affected by the change]. The safety significance of the change is considered following the first determination, i.e., does the change involve an unreviewed safety question. The licensee's NAP-59 procedure correctly identifies this two step 10 CFR 50.59 process. However, the response to the question incorrectly suggests that the questions "Does the proposal change the facility as described in the SAR?", and "Can this change impact the safe operation of the plant?" are the same, and consequently may lead to improper determination of 10 CFR 50.59 applicability.

D. Implementation Review

Hope Creek

Fourteen completed 10 CFR 50.59 reviews and safety evaluations were reviewed by the Hope Creek PM. This represents about 5% of all the 10 CFR 50.59 reviews and SEs that were completed between August 1991 and December 1992, as documented by the licensee in their monthly operating reports. Additionally, thirty-two completed 10 CFR 50.59 applicability reviews were inspected, i.e., items for which the licensee determined that 10 CFR 50.59 did not apply and required no SE. The sample was drawn from a list of items provided by the licensee and generally covered calendar year 1992.

There were no safety significant problems that were identified during the inspection, for reviews and safety evaluations that were completed under 10 CFR 50.59 requirements. However, there were instances when the licensee did not follow its own procedure. For example:

- NAP-59 paragraph 5.1.1 states in part "The description [of the CTE] shall be specific and unambiguous. It shall also include a discussion of the applicable design, operation and regulatory requirements that relate to the proposal." Contrary to this requirement, there were three 10 CFR 50.59 reviews and SEs that did not contain an adequate description of the change. For example, design change package (DCP) 4EC-3111 Package 4, does not include a discussion of the design, operation or regulatory requirements that relate to the DCP. Other reviews and SEs that did not have an adequate description of the change were: DCPs 4HX-0331 and 4EC-3002, package 1.

- NAP-59 paragraph 4.7 states "The 10 CFR 50.59 Review and Safety Evaluation shall address all phases of the change, test or experiment, including the installation, removal and testing phases...." Five of the 10 CFR 50.59 reviews and safety evaluations for DCPs and Temporary Modifications (T-mods) that were reviewed by the inspector did not contain this required discussion. For example, TMR 91-046 does not contain any discussion of the installation or testing phases for this T-mod; and 4EC-3226 does not contain a discussion of the testing phase for this DCP. Other DCPs and T-mods that did not have an adequate discussion of all phases of the CTE were: DCPs 4HX-3342 and 4EC-3182, package 9; and T-mod TMR 92-020.

Similar in nature to the 10 CFR 50.59 reviews and SEs discussed above, there were instances when the licensee did not follow its own procedure relative to items that were reviewed for 10 CFR 50.59 applicability, but did not require an SE (i.e., the licensee determined that 10 CFR 50.59 was not applicable). For example:

- The entire description for Revision 4 to procedure HC.IC-DC.ZZ-070 states "The proposed procedure revision rewrites the procedure to bring it in accordance with the vendor recommended method of testing and calibration." This description is not specific and it does not contain any discussion of the applicable design, operation and regulatory requirements. Additionally, Revision 5 to procedure HC.IC-TR.AB-001(Q) does not contain a discussion of the applicable design, operation and regulatory requirements.

- NAP-59 paragraph 5.2.2 states in part "The 10 CFR 50.59 Review shall set forth the SAR sections reviewed, and the basis used in making the determination. A simple statement of conclusion is not sufficient, nor is merely restating the question in the form of an answer. The level of detail must be sufficient to allow an independent reviewer to verify the conclusion, and to permit review by external organizations (i.e., the NRC)." Revision 3 to procedure HC.OP-AP.ZZ-0111(Q) does not reference the specific SAR sections reviewed. DCP 4HC-339, Package 1, states that "UFSAR [i.e., updated FSAR] Section 10.4.4 was reviewed and it is determined that the SAR is not affected by these modifications." However, the inspector's review determined that UFSAR Section 10.4.4 does not apply to the system involved in the DCP. The UFSAR Section referenced should have been 10.4.5.

The procedure compliance discrepancies noted above are not safety significant by themselves. However, they indicate that the licensee is being less critical in this area than is required. The licensee was previously informed of similar NAP-59 procedure compliance discrepancies in inspection report 50-272/91-26, 50-311/91-26 and 50-354/91-19. Since NAP-59 was written to implement the requirements of 10 CFR 50.59, deviations from the guidance in NAP-59 could lead to a violation of 10 CFR 50.59.

For example, NAP-59 paragraph 6.2 defines changes in the facility as described in the SAR as "...modifications that affect the design, function or method of performing the function of a structure, system or component described in the SAR. These changes are not limited to structures, systems or components specifically described in the SAR, since changes to components not specifically described in the SAR can affect the design or operation of systems or components that are described in the SAR.". Paragraph 6.2.1 further states that changes include "...Operation with known setpoint drift or degradation of equipment due to creep, fatigue, corrosion, or erosion." Notwithstanding these specifications, the following is an example of a case that was improperly screened from the need to perform a SE:

- Relative to DR HTE 92-230, the licensee supported a "use-as-is" disposition for unqualified gauges in the gland seal portion of the high pressure coolant injection (HPCI) system. In the 10 CFR 50.59 review, the licensee states, "The pressure gauges are not described [in the UFSAR]." The PM determined that this statement is incorrect. These gauges are described in UFSAR Figure 6.3-2 as being within the "Q" boundary. Furthermore, in order to resolve this DR, the licensee changed the normal position of the isolation valves for these gauges from open to closed. However, UFSAR Figure 6.3-2 clearly shows the isolation valves for these gauges as being normally open. Since the licensee changed the facility as described in the SAR, a 10 CFR 50.59 SE should have been performed.

This example constitutes a violation of 10 CFR 50.59(b)(1), which states, in part, that records of changes to the facility as described in the UFSAR "must include a written safety evaluation which provides the basis for the determination that the change, test, or experiment does not involve an unreviewed safety question." In addition, the Technical Specifications (TS), Section 6.5.1.6.e. and Section 6.5.2.4.2.a. requires the Station Operations Review Committee (SORC) and the Offsite Safety Review Group (OSR), respectively, to review all safety evaluations completed under the provisions of 10 CFR 50.59. Because the licensee determined that 10 CFR 50.59 did not apply to this change, a safety evaluation was not prepared. Therefore, a SORC and OSR review was not performed as required by the TS. (Section 8.3.1 pertains to the apparent violation.)

Salem

Twenty-two completed 10 CFR 50.59 reviews and safety evaluations (SEs) were reviewed by the project manager (PM). This represents about 5% of all the 10 CFR 50.59 reviews and SEs that were completed between July 1991 and December 1992, as documented by the licensee in their monthly operating reports. Additionally, thirty-six completed 10 CFR 50.59

reviews were inspected ,i.e., items for which the licensee determined that 10 CFR 50.59 did not apply and required no SE. The sample was drawn from a list of items provided by the licensee and generally covered calendar year 1992.

For reviews and safety evaluations that were completed under 10 CFR 50.59 requirements, one 10 CFR 50.59 procedure review relative to NC.NA-AP.ZZ-0036(Q), "Control of Information System and Computer Resources," did not meet the licensee's procedural requirement that sufficient detail be included to allow a reviewer to independently arrive at the same conclusion.

The following comments concern items that were reviewed under 10 CFR 50.59, but did not required a 10 CFR 50.59 SE (i.e., the licensee determined that 10 CFR 50.59 was not applicable to these items):

- Deficiency Report 92-024 addressed coating of the 22 RHR Pump Room Cooler tubesheet, but was not complete in that it was not a stand-alone document. The method of repair, which represented the change being made was not discussed and the engineering evaluation that addressed the issue was not referenced in the 10 CFR 50.59 Review.

- Deficiency Report 92-644 addressed the deficiencies found during testing of valve 1SJ135, but contained errors in the evaluation in that it indicated that the maximum calculated torque exceeded the continuous duty torque by 13% It actually exceeded the torque by 30.8% The incorrect maximum calculated thrust value was used throughout the evaluation. Subsequent to this finding, PSE&G identified the following to the PM:

    a. The use of the maximum calculated thrust in the evaluation is being reconsidered. Either the measured or calculated value of thrust will be used in a revision of the deficiency report, depending on which value is most conservative.

    b. The calculation of the amount the torque exceeded the continuous duty torque should have been 30.8%. This will be corrected in the revised deficiency report.

In addition, there were two incorrect 10 CFR 50.59 applicability determinations. These were both temporary modifications (T-mods). The T-mods were TMR 92-031 that provided instructions for disconnecting the normal power (vital bus 1B) and reconnecting a temporary power supply (vital bus 1C) to the No. 12 Auxiliary Building supply fan; and TMR 92-043 that installed a blank flange in the service water system. The following pertains:

- TMR 92-031 was evaluated by the licensee as not changing the Updated Final Safety Analysis Report (UFSAR). The licensee's 10 CFR 50.59 review stated that the reasons that this T-mod did not change the UFSAR were: The fan would be inoperable during the "1B" bus outage and this T-mod would provide temporary power to make the fan operable. The function of the fan remains unchanged, therefore, this T-mod is not a change to the SAR. However, the PM found that the Auxiliary Building Ventilation System and the vital bus connection of the supply fans are included in Tables 8.3-2 and 8.3-3, and Figure 8.3-4A in the UFSAR. In addition, TMR 92-031 referenced TMR-006 for a discussion of the separations requirements which did require a 10 CFR 50.59 Safety Evaluation.

- TMR 92-043 was evaluated as not changing the UFSAR because the blank flange served to isolate portions of the service water header that were in service. There is a manual valve (22SW414) installed in the system and the blank flange was installed upstream of that valve to ensure positive (leak tight) isolation. However, this changed the system as shown on Figure 9.2-1B in the UFSAR.

The failure of the licensee to identify the two T-mods, described above, as changes to the UFSAR constitutes a violation of 10 CFR 50.59(b)(1), which states in part that records of changes to the facility as described in the UFSAR "must include a written safety evaluation which provides the basis for the determination that the change, test, or experiment does not involve an unreviewed safety question." In addition, the Technical Specifications (TS), Section 6.5.1.6.e. and Section 6.5.2.4.2.a. requires the Station Operations Review Committee (SORC) and the Offsite Safety Review Group (OSR), respectively, to review all safety evaluations completed under the provisions of 10 CFR 50.59. Because the licensee determined that 10 CFR 50.59 did not apply to these changes, a safety evaluation was not prepared. Therefore, SORC and OSR reviews were not performed as required by the TS. (Section 8.3.1 pertains to the apparent violation)

8.3.1 Apparent Violation

The safety significance of the indicated apparent violations (as detailed in Section 8.3 D. relative to Hope Creek and Salem) is low. However, since there were several discrepancies found in a relatively small sample size, in aggregate, these findings indicate a weakness in the licensee's implementation of the 10 CFR 50.59 program, and are considered as examples of an apparent violation of the requirements of 10 CFR 50.59 (VIO 50-354/93-06-03; VIO 50-272 and 50-21/93-08-03).

9. LICENSEE EVENT REPORTS (LER), PERIODIC AND SPECIAL REPORTS, AND OPEN ITEM FOLLOW-UP

9.1 LERs and Reports

PSE&G submitted and reviewed for accuracy and evaluation adequacy the following special and periodic reports.

- Salem and Hope Creek Monthly Operating Reports for March 1993.
- Salem and Hope Creek Annual Personnel Exposure and Monitoring Report for 1992.
- Salem Unit 2 Special Report 93-1 regarding the inoperability of radiation monitors 2R45B and 2R45C.
- Hope Creek 1992 Annual Environmental Operating Report.

The inspector concluded that the licensee appropriately issued the above reports.

Salem LERs

Unit 1

- LER 92-26-02 is a supplemental LER which addressed three additional events (radiation monitoring system ESF actuations) which had the same root cause (increased containment activity) as the first event. The inspector monitored the licensee's efforts in this area, and closed this LER.
- LER 93-04 discussed an automatic reactor trip from 100% power due to an equipment failure (overtemperature differential temperature gain selector switch). The inspector reviewed this event in NRC Inspection 50-272/93-02, and closed this LER.
- LER 93-05 concerned a reactor protection system actuation (reactor/turbine trip signal) while in Mode 3 (Hot Standby) due to personnel error. The inspector reviewed this event as described in NRC Inspection 50-272/93-02, and closed this LER.
- LER 93-06 described a Technical Specification required shutdown due to the loss of one offsite transmission network. The inspector reviewed this event in NRC Inspection 50-272/93-02, and closed this LER.
- LER 93-07 concerned two Technical Specification (TS) 3.0.3 entries more than one analog rod position indicator (ARPI) per bank became inoperable. Actual control rod positions were subsequently verified for the associated ARPIs. For each occasion, TS 3.0.3 was exited within one hour. The inspector noted that the licensee's investigation and corrective actions were appropriate, and closed this LER.

- LER 93-08 discussed a design concern associated with control air containment isolation valves. See Section 7.1.A of this report for details. The inspector closed this LER.

- LER 93-09 described a Technical Specification 3.0.3 entry due to a failed boric acid storage tank level indication. The inspector reviewed this event in NRC Inspection 50-272/93-02, and closed this LER.

Unit 2

- LER 93-05 (See Section 2.1.B). This LER is closed.

Hope Creek

- LER 93-01 (See Section 4.3.2.A). This LER remains open.

9.2 Open Items

The inspector reviewed the following previous inspection items during this inspection. These items are tabulated below for cross reference purposes.

Site	Report Section	Status
Salem		
272&311/91-16-01	4.3.1.B	Closed
272&311/92-01-05	8.1.B	Closed
Hope Creek		
354/92-03-04	7.2.A	Closed

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 NRC 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
3/29-4/2/93	Inservice Inspection	50-272&311/93-09	McBrearty
4/5-9/93	Radiological Controls	50-272&311/93-10	Nimitz
4/5-7/93	Security	50-272&311/93-11; 50-354/93-07	Albert

ATTACHMENT 1

50.59 EVALUATIONS AND SCREENED OUT PACKAGES REVIEWED

HOPE CREEK

A. DESIGN CHANGE PACKAGES

1.	4EC-3226	Modified the logic of the E and F Filtration Recirculation Ventilation System recirculation fans. **(From Mar 92 Hope Creek Monthly Operating Report (MOR))**
2.	4EC-3342	Added time delay into the closing circuit of the alternate infeed breaker in the slow and dead bus transfer schemes to prevent the alternate infeed from closing too soon and to enable the sequencer to reset after a bus transfer. **(From Apr 92 MOR)**
3.	4HX-0331/01	Replaced mechanical snubbers with hydraulic snubbers. **(From Aug 92 MOR)**
4.	4EC-3002/01	This DCP replaced 2" schedule 80 pipe with schedule 40 pipe. **(From Dec 92 MOR)**
5.	4EC-3111/04	This DCP diverted Service Water from the Cooling Tower Basin and Cooling Tower Bypass Line to a manhole in the yard. **(From Dec 92 MOR)**
6.	4EC-3182/09	This DCP changed the power supply short circuit protection of field wires on 1E instrument loops by replacing fuses with resistors. **(From Dec 92 MOR)**

B. PROCEDURES

1.	NC.NA-AP.ZZ-0071(Q)	Revision 0 - Describes a zero defect fuel performance program that will prevent or mitigate the impact of failed fuel on plant operations. The procedure was developed to satisfy the recommendations of INPO SOER 90-02, "Nuclear Fuel Defects." **(From Aug 91 MOR)**
2.	HC.IC-LC.AE-0005(Q)	Revision 0 - This procedure installs jumpers to bypass the 20% total feedwater flow interlock to the recirculation pump speed limiter to preclude an actual runback from occurring during the transmitter calibration. **(From Mar 92 MOR)**

3. HC.SA-AP.ZZ-0052(Q) Revision 7 - Provides guidance for the station departments involved in ensuring that water chemistry parameters are maintained in accordance with the appropriate vendor and industry guidelines. **(From Aug 92 MOR)**

C. TEMPORARY MODIFICATIONS (T-mods)

1. 91-046 Modified the circuit for the measurement of river water temperature. Three of four temperature detectors are currently providing unreliable readings. **(From Sep 91 MOR)**

2. 92-020 Installed Control Air tubing between a pressure control valve in the Gaseous Radwaste system and its associated instrumentation. **(From Aug 92 MOR)**

D. UFSAR CHANGES AND DEFICIENCY REPORTS

1. 6.2.4.4.3 Assigns total observed leakage through the outboard MSIV only when leak rate testing is performed between the MSIV and the MSSV. **(From Jan/Feb 92 MOR)**

2. HMD 92-009 Addresses a through-wall leak on a Station Service Water instrument line. **(From Jan/Feb 92 MOR)**

3. HTE 92-010 Addresses the installation of schedule 40 pipe instead of schedule 80 pipe at several SSW 1" and 1.5" root valve lines. **(From Oct 92 MOR)**

E. SCREENED OUT ITEMS

DCPs

1. 4HC-339, pkg 1 Replaces existing NaOCl storage tanks with Durakane 411 lined tanks for the Circulating Water Hypochlorination System.

2. 4EC-3046 Refurbishes two 500KV Type SFA gas circuit breakers.

3. 4HE-0001 Modifies the Reactor Core Isolation Cooling (RCIC) system flow controller setpoint raise and lower circuit.

4.	4HE-0002,pkg 1	Custom fits a new hinge and disc to the seat inside the inboard feedwater containment isolation valve 1AEV-003.
5.	4HE-0013	Installs carbon steel angles along the top and bottom of the FRVS straightening vanes.

Procedures

1.	HC.CH-GP.ZZ-0006(Q)	Revision 0 - Provides guidance to the Chemistry Department in the event of a SCRAM.
2.	HC.CH-EO.SH-0004(Q)	Revision 6 - Deletes steps that have been integrated into HC.CH-EO.SH-0005(Q).
3.	HC.IC-CC.AB-041(Q)	Revision 15 - Incorporates new setpoint values for channels A, C, E and J.
4.	HC.IC-DC.ZZ-070	Revision 4 - Rewrites the procedure to bring it in accordance with the vendor recommended method of testing and calibration.
5.	HC.IC-LC.FC-001(Q)	Revision 2 - Incorporates two technical changes.
6.	HC.IC-SC.BH-002(Q)	Revision 0 - Created to test and calibrate the standby liquid control system storage tank level transmitters.
7.	HC.IC-TR.AB-001(Q)	Revision 5 - Changes the total time response acceptance criteria.
8.	HC.MD-AP.ZZ-0014(Q)	Revision 7 - Changed format to comply with guidelines of NC.NA-AP.ZZ-0032(Q).
9.	HC.MD-GP.ZZ-0021(Q)	Revision 3 - Various changes were accomplished by this revision.
10.	HC.MD-PM.KJ-005(Q)	Revision 5 - Revised procedure for biennial review.
11.	HC.MD-ST.KE-001(Q)	Revision 9 - Incorporated changes that were required by the implementation of DCP 4EC-1043.
12.	HC.OP-AP.ZZ-0111(Q)	Revision 3 - Revised procedure for biennial review.

T-mods

1.	92-005	Provided power feed to UPS load disconnect switch.
2.	92-015	Installs pressure and flow transmitters to components 1AEPDT-N002A/N002B and 1APT-3686A/3686B in support of a Unit Heat Rate Evaluation.
3.	92-025	Allows use of polar crane auxiliary hoist while the main hoist is de-energized for maintenance.
4.	92-033	Installs a temporary transformer in panel 1B-C-156.
5.	92-034	Abandons 11 LPRM cables and subsequent temporary routing of additional cables.
6.	92-035	Addresses the use of a silver bronze pressure sensing tube in lieu of stainless steel.

DRs

1.	HTE-92-122	Dispositions the condition of the backwash outlet flanges on the C Service Water strainer 1C-F-509.
2.	HTE-92-124	Addresses the damaged concrete lining on Service Water discharge line EA-24"-STJ-002.
3.	HTE-92-148	Addresses the presence of material anomalies on the Reactor Feed pump anti-vortex dam.
4.	HTE-92-230	Supports use-as-is disposition for pressure gauges in the gland seal portion of the High Pressure Coolant Injection (HPCI) system.
5.	HTE-92-232	Supports use-as-is disposition for Control Rod Drive (CRD) 3015 not meeting the acceptance criteria for friction testing identified in procedure HC.OP-FT.BF-0004(Q), Revision 2.
6.	HMD-92-159	Addresses the repair of the seating surface of the disc to testable swing check valve 1BCV-033.
7.	HMD-92-176	Restored HPCI turbine shaft gland seal area to acceptable surface finish.

8.	HMD-92-250	Repaired crack in FRVS flow straightening vane by drilling a hole at the end of the crack.
9.	HIC-92-202	Repaired LPRM detector cable outer jacket tear.

SALEM

A. DESIGN CHANGE PACKAGES

1.	1EC-3205	RVLIS Refueling. (From Unit 1 Dec. 92 MOR)
2.	1EC-3195, Pkg 1	Lube Oil Storage Facility Revitalization Project FC-0001 Units 1 and 2. (From Unit 1 Oct. 92 MOR)
3.	1EC-3186, Pkg. 1	Steam Generator Feed Pump High Discharge Pressure Trip. (From Unit 1 July 92 MOR)
4.	1SC-2267 Pkg. 2	SEC Containment Spray Actuation. (From Unit 1 May 92 MOR)
5.	1EC-3162 Pkg 1	Installation of Turbine Auto Stop Oil System Filters. (From Unit 1 April 92 MOR)
6.	2EC-3110 Pkg 1	Allowable Value and Setpoint for Containment Hi-Hi Pressure. (From Unit 2 March 92 MOR)
7.	2EC-3087 Pkg 1	RHR Monitoring During Mid-Loop Operations. (From Unit 2 Feb. 92 MOR)
8.	2SC-2267 Pkg 1	Safeguards Equipment Cabinet Control Electronics Unit Replacement, Revision 1. (From Unit 2 Jan. 92 MOR)

B. PROCEDURES

1.	NC.NA-AP.ZZ-0036 (Q)	Control of Information System and Computer Resources. (From Unit 1 Nov. 92 MOR)
2.	NC.NC-AP.ZZ-0013 (Q)	Control of Temporary Mods, Revision 1. (From Unit 1 April 92 MOR)
3.	S1.OP.AB.ROD-0004 (Q)	Rod position Indicator Failure. (From Unit 1 Jan. 92 MOR)

4.	S1.OP-SO.RC-0005 (Q)	Draining the RCS, Revision 2. (From Unit 1 July 92 MOR)
5.	TSI.OP-SO.AF-0001 (Q)	Aux Feed Operation. (From Unit 1 June 92 MOR)

C. TEMPORARY MODIFICATIONS

1.	92-057	Installation of Temporary Air Dryer. (From Unit 1 Sept. 92 MOR)
2.	TMR 92-015	Removing/Returning 2A 125VDC Bus From/To Service. (From Unit 2 Feb. 92 MOR)
3.	TMR 92-037	Monitoring Temperatures Inside Pressurizer Enclosure. From Unit 1 June 92 MOR)

D. SAFETY EVALUATIONS, DEFICIENCIES, SAR CHANGES, AND TECH SPEC INTERPRETATIONS

1.	S-O-AF-MSE-0812	Potential Cavitation of the Auxiliary Feedwater Pumps. (Safety Evaluation) (From Unit 1 Sept. 92 MOR)
2.	S/E WO 920417117	Breaching a Penetration Seal. (Safety Evaluation) (From Unit 2 May 92 MOR)
3.	SMD-92-735	12 Service Water Return From 12 CCHX Wall Thinning. (DR) (From Unit 1 Sept 92 MOR)
4.	DR SMD 92-262	Primary Water Storage Tank. (DR) (From Unit 1 May 92 MOR)
5.	SCN# 92-42	Updating SAR - Control Room Habitability. (SAR change) (From July 92 MOR)
6.	TSI #3.7.1.1	Operation of Salem Units 1 and 2 with Reduced Main Steam Safety Valve Flows (Tech Spec Interpretation) (From Unit 1 July 92 MOR)

E. SCREENED OUT ITEMS

1. Temporary Modifications

   a. 92-071, Removal of Reverse Power Relay (Salem 2).

b. 92-031, Jumpers and Lifted Leads to Supply Temporary Power during 1B Bus Outage.

c. 92-017, Clamp on Orifice and Seal of 13MS200.

d. 92-026, Jumpers and Lifted Leads to Supply Temporary Power to #12 Spent Fuel Pool pump during Bus 1B Outage

e. 92-006, Jumpers and Lifted Leads to Supply Temporary Power to #11 Spent Fuel Pool pump during Bus 1C Outage

f. 92-029, Removal of Manipulator Crane West Trolley Limit of Travel Bumper

g. 92-043, Isolation of 22 Service Water Chiller Header

2. <u>Deficiency Reports</u>

a. SMD 93-012, CFCU Inlet/Outlet Flange Repair.

b. SMD 92-708, Unit 2 Reactor Trip Breaker Roller Assembly Out-of-Specification.

c. SMD 92-532, ISI - 13 Steam Generator Object Removal.

d. SMD 92-664, Evaluation of 11SJ40 Closing Thrust

e. SMD 92-644, Measured Thrust for 1SJ135 Higher than Maximum

f. SMD 92-615, Thrust for 1CV116 Lower than Required

g. SMD 92-182, Airlock Door Hinge Pin Indications

h. SMD 92-024, 22RHR Pump Room Cooler Tubesheet Corrosion

i. SMD 92-546, Indication of Pipe to Valve Weld (12MS167)

j. SMD 92-362, No Limiter Plate on 1CS16 Torque Switch

k. SMD 92-261, Spring Can Hanger, Unable to Adjust

3. <u>Procedures</u>

a. SC.DE-AP.ZZ-0055 (Q), Detailed Procedure for E/C Monitoring Program.

b. SC.RC-TI.ZZ-0190 (Q), Software Control

c. S2.OP-SO.PZR-0003 (Q), Pressurizer Relief Tank Operation

d. S2.RE-RA.ZZ-0008 (Q), Post Refueling Initial Criticality

e. 2-II-8.3.4, Draining the Reactor Refueling Cavity

f. 1IC-14.3.002, Response Time Testing

g. S1.OP-SO.WG-0008, Discharge of No. 11 Waste Decay Tank to the Plant Vent

h. II-15.3.2, Containment Entry

i. SC.MD-GP.ZZ-0022, Torquing of Fasteners

j. S1.IC-CC.RM-0064 (Q), Plant Vent Radiation Monitor Channel Calibration Procedures

k. 2IC-4.5.060, Calibration of Radiation Monitors

l. S2.RE-RA.22-0002 (Q), Inverse Count Rate Ratio During Control Rod Withdrawal

4. <u>Design Change Packages</u>

a. 2EC-3154, Change to the AMSAC Diagnostic Software

b. 2EC-3150/1, Change the Circuit Breakers for the Vacuum Pumps

c. 2EC-3137, Change the Circulating Water Intake Screen Wash Strainer Motor Circuit Breaker

d. 1EC-3200, Install a Hot Water Heater in the Turbine Building

e. 1SC-2269, Install Cable and Raceways to Support Salem Electrical Distribution Project

f. 2EC-3085/1, Changes Protective Relays for Main Generator Flashover Protection