U. S. NUCLEAR REGULATORY COMMISSION

REGION I

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Report No. 50-354/87-23

Docket 50-354

License NPF-57

Licensee: Public Service Electric and Gas Company

Facility: Hope Creek Generating Station

Conducted: October 27, 1987 - November 30, 1987

Inspectors: R. W. Borchardt, Senior Resident Inspector
D. K. Allsopp, Resident Inspector
R. R. Brady, Reactor Engineer

Approved: P. Swetland 1/14/88
P. Swetland, Chief, Projects Section 2B Date

Inspection Summary:
Inspection on October 27, 1987 - November 30, 1987 (Inspection Report Number 50-354/87-23)

Areas Inspected: Routine onsite resident inspection of the following areas: followup on outstanding inspection items, operational safety verification, surveillance testing, maintenance activities, engineered safety feature system walkdown, unauthorized jumpers, preparation for refueling, and licensee event report followup. This inspection involved 249 hours by the inspectors.

Results: A violation of station procedures relating to the review, authorization and control of temporary modifications is cited in this report (paragraph 7).

Our review of your preparations for the refueling outage and our observations of new fuel receipt inspections found these activities to be well controlled.

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Details

1. Persons Contacted

   Within this report period, interviews and discussions were conducted with Mr. S. LaBruna and members of the licensee management and staff and various contractor personnel as necessary to support inspection activity.

2. Followup on Outstanding Inspection Items

   a. (Closed) Inspector Follow Item (86-56-01); Several deficiencies in the licensee's cold weather preparation program were identified. The inspector reviewed design change package (DCP) 4-TMJ-86-264 which installed a control room overhead annunciator to monitor the hydrogen-oxygen (H2O2) analyzer heat tracing panels. A failure of the H2O2 heat trace system (power failure, current failure, or low heat trace temperature) will cause a control room annunciator to alarm as verified by functional testing conducted after DCP implementation. The licensee developed a heat trace system training course and an H2O2 analyzer local alarm response book which were both reviewed by the inspector. The inspector has determined during discussions with equipment operators (EO) that the heat trace system training course has been effective in providing EOs with a satisfactory understanding of heat trace control systems. The licensee has completed a 100% verification of agreement between heat trace controller setpoints and instrument calibration data card settings. The inspector performed random spot checks of heat trace controller setpoints and found no discrepancies. This item is closed.

   b. (Closed) Deviation (87-14-01); Contrary to an FSAR commitment, there was no control room alarm for an H2O2 analyzer sample line heat trace failure. The inspector reviewed the licensee's response to this deviation dated August 6, 1987. As discussed in the above closed item, this control room alarm has been installed and functionally tested. This item is closed.

   c. (Closed) Inspector Follow Item (87-14-02); Several "B" H2O2 local heat trace circuits (CN3, CN4, CN5) appeared to be malfunctioning in that they were almost constantly in alarm and periodically could not be reset. The licensee has determined the nitrogen (N2) makeup paths to the drywell utilize portions of the "B" H2O2 sample lines. Thus, whenever N2 makeup to the drywell is in progress, the H2O2 sample lines cool off and actuate the low temperature alarm. The licensee has conducted equipment operator and control room operator training to ensure operators anticipate this alarm and properly reset it. Both local H2O2 analyzer heat trace panels have alarm response books to ensure correct operator response. Heater circuit CN3 had a mechanical controller fault which has been corrected and the controller is working properly. This item is closed.

3. Operational Safety Verification

3.1 Inspection Activities

On a daily basis throughout the report period, inspections were conducted to verify that the facility was operated safely and in conformance with regulatory requirements. The licensee's management control system was evaluated by direct observation of activities, tours of the facility, interviews and discussions with licensee personnel, independent verification of safety system status and limiting conditions for operation, and review of facility records. The licensee's adherence to the radiological protection and security programs was also verified on a periodic basis. These inspection activities were conducted in accordance with NRC inspection procedures 71707, 71709 and 71881 and included weekend and backshift inspections conducted on November 7, November 24, and November 25, 1987.

3.2 Inspection Findings and Significant Plant Events

The unit entered this report period at approximately 10% power while completing operational verification checks of the new phase "A" main transformer. The transformer was installed due to the extensive damage resulting from a transformer explosion and fire on October 13, 1987. All checks were completed satisfactorily and the unit proceeded to 100% power. The unit remained at essentially full power for the entire inspection period with the exception of short duration power reductions for maintenance or surveillance activities, and as discussed below.

On November 5, 1987, a mechanical failure of the B turbine building chiller resulted in the tripping of all chilled water pumps in service. Freon leaking from failed condenser tubes accumulated in the chilled water pump casings and caused the pumps to trip on low flow. Drywell cooling was cross tied to the reactor auxiliary cooling system (RACS) and reactor power was reduced to 90%. These actions resulted in limiting drywell temperatures to approximately 113 degrees F and drywell pressure to 0.8 psig. Both of these values are well within the technical specification limits of 135 degrees F and 1.5 psig, respectively. The chilled water system was vented and returned to service on the same day. Drywell cooling was returned to the chilled water system and reactor power restored to 100%. An inspector was in the control room throughout this event and noted that operator response was controlled and well thought out. The inspector has no further questions relating to this event.

On November 10, 1987, the Topaz inverter that supplies channel C emergency core cooling system (ECCS) logic control power was de-energized due to a high voltage condition. Electricians were performing a high voltage trip test on one of the two redundant channel "C" 125 VDC battery chargers when the second charger became

unstable. The resulting electrical transient tripped both battery chargers and the "C" Topaz inverter on high voltage. Although the ECCS logic and the battery chargers were quickly re-energized, this event identified a weakness in the battery charger test procedure. There appeared to be no reason that the charger under testing cannot be isolated and disconnected from the 125 VDC bus since each battery charger is 100% capacity. The licensee is currently evaluating a change to the test procedure and the inspector will followup on the procedure revision in a future inspection. The licensee is also reevaluating the high voltage trip setpoints of the battery chargers and the Topaz inverters for all four 1E busses.

On November 23, 1987 at 10:30 a.m., a control building isolation occurred due to an I&C technician personnel error. The technician was performing a surveillance test on the refueling floor exhaust radiation monitor circuit when he accidentally adjusted a setpoint on the control room radiation monitor circuit and caused the isolation. The isolation was immediately reset and ventilation systems were returned to normal.

During this report period, the plant was inspected for cold weather preparation. Plant conditions were found to be satisfactory, however, 2 areas require followup during the next inspection period. The inspector noted that the condensate storage tank (CST) overflow line does not have complete coverage with heat tracing or insulation, although the overflow line is heat traced and insulated from the CST connection to about five feet above where the line enters the ground. The licensee is researching the design drawings to determine the heat trace requirements for this overflow pipe. The inspector also found that ten of twenty-one heat trace control panels are in alarm in the control room. These heat trace control panels are not safety related and supply components such as liquid radioactive waste. The licensee has initiated work requests to troubleshoot these heat trace control panel alarms. The inspector will follow up on these items in the next inspection report.

No violations were identified.

4. Surveillance Testing

4.1 Inspection Activity

During this inspection period, the inspector performed detailed technical procedure reviews, witnessed in-progress surveillance testing, and reviewed completed surveillance packages. The inspector verified that the surveillance tests were performed in accordance with Technical Specifications, licensee 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:

- IC-FT.BC-005 RHR Discharge Flow
- IC-FT.BB-023 RPV High Pressure
- IC-FT.SK-040 RWCU Steam Leak Detection Logic
- IC-FT.SP-052 "B" Main Steam Line Radiation Monitor
- IC-FT.SE-020 Channel B Rod Block Monitor

No violations were identified.

5. Maintenance Activities

5.1 Inspection Activity

During this inspection period the inspector 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.

5.2 Inspection Findings

Portions of the following activities were observed by the inspector:

Work Order	Procedure	Description
871102154	CH-DC.ZZ-0002	Repair, replace, and recalibrate "B" H2O2 analyzer sample pump.
871118097	CJP-H87-092	Repair leak on service water pipe to safety auxiliary cooling heat exchanger.

On November 17, 1987, an equipment operator was in the safety auxiliary cooling system (SACS) heat exchanger room when a leak started from the A2 heat exchanger inlet pipe. The leak was determined to be from a 3/8 inch diameter hole in the service water supply to the SACS heat exchanger. The heat exchanger was isolated and an investigation commenced to determine the extent of pipe damage. This examination determined that the pipe wall thickness degradation was extremely localized and that even immediately adjacent to the hole the wall thickness was significantly greater than minimum wall requirements. The repair method consisted

of welding a 3/4 inch plug onto the 26 inch service water pipe. The inspector observed in process work activities and reviewed the job code package and found conditions to be acceptable. The subject piece of pipe will be replaced during the upcoming refueling outage.

No violations were identified.

6. Engineered Safety Feature (ESF) System Walkdown

6.1 Inspection Activity

The inspectors independently verified the operability of selected ESF systems by performing a walkdown of accessible portions of the system to confirm that system lineup procedures match plant drawings and the as-built configuration. This ESF system walkdown was also conducted to identify equipment conditions that might degrade performance, tc determine that instrumentation is calibrated and functioning, and to verify that valves are properly positioned and locked as appropriate. This inspection was conducted in accordance with NRC inspection procedure 71710.

6.2 Inspection Findings

The high pressure coolant injection (HPCI) system was inspected and conditions were found to be generally satisfactory. A number of minor material and housekeeping deficiencies were noted, however, none were observed that adversely affected the safety related function of the HPCI system. The majority of these deficiencies had been previously identified by the licensee and were being tracked for correction.

In addition to in-field verifications of system operability, the inspector conducted an in-depth review of the following surveillance test procedures:

OP-ST.BJ-001	HPCI System Piping and Flow Path Verification - Monthly
OP-ST.BJ-002	HPCI System Functional Test - 18 Month
OP-ST.BJ-003	HPCI System Functional Test (Low Pressure) - 18 Month
OP-IS.BJ-001	HPCI Valves In Service Test
IC-CC.BJ-005	E41-N661A, HPCI Condensate Storage Tank Level
IC-CC.BJ-006	E41-N661E, HPCI Condensate Storage Tank Level

All procedures were found to adequately fulfill the testing requirements of technical specifications.

No violations were identified.

7. Unauthorized Modifications

On November 10, 1987, an I&C technician who was troubleshooting a channel "D" Source Range Monitor (SRM) upscale/inoperative alarm found a test relay installed in the circuit that was not authorized by any procedure or the station temporary modification program. The licensee's investigation determined that this test relay was installed by a technician on November 9, 1987, while performing a surveillance test on SRM "A". The test relay was installed in order to allow testing of an annunciator circuit, however, the technician failed to adhere to station and maintenance department procedures by installing what amounted to a temporary modification without proper authorization and review.

On November 19, 1987, an unauthorized jumper was discovered in the circuit for control room overhead annunciator E5-D2 "Control Area Chilled Water Trouble". The licensee's investigation of this discrepancy determined that the control room operators had directed the shift I&C technician to install a jumper in order to disable an input to the annunciator circuit. Scheduled maintenance on a control area chiller was causing this annunciator to repeatedly go into and come out of alarm. The alarm was in turn creating a nuisance and distraction to the operators. Although the installation of this jumper may have been warranted, station controls for review, authorization and control of temporary modifications were completely bypassed.

Station Administrative Procedure SA-AP.ZZ-13(Q), "Control of Temporary Modifications" defines a temporary modification as "a modification to a permanent plant component or system that alters the approved design configuration. This includes lifted leads, jumpers, mechanical modifications, and electrical modifications..." This administrative procedure also states that all modifications made in accordance with this procedure shall be reviewed and approved prior to installation except where specifically authorized by a maintenance or surveillance procedure. The safety impact of these unauthorized modifications was minor and the licensee promptly removed them using proper controls. The installation of unauthorized temporary modifications constitutes an apparent violation of station procedures. (50-354/87-23-01)

8. Refueling Preparations and Fuel Receipt Inspections

During this inspection a review of the procedures and administrative controls used to perform a refueling was conducted. This inspection included a procedure review of administrative controls as well as technical procedures used to obtain and maintain specific plant conditions; observation of new fuel inspection qualification training; and observations of refueling preparation surveillances, and receipt and storage of new fuel.

8.1 Procedure Review

In the following areas procedures were reviewed for technical adequacy and compliance with ANSI 18.7-1976.

8.1.1 Receipt, Inspection, and Storage of New Fuel

The procedures that outline departmental responsibilities, interfaces, and technical requirements to receive, inspect, and store new fuel shipments were reviewed. These included:

- SA-AP.ZZ-039 "RECEIPT OF NEW FUEL"
- RE-FR.ZZ-004 "NEW FUEL INSPECTION"
- MD-FR.KE-008 "NEW FUEL HANDLING AND STORAGE"

These procedures provide adequate control of new fuel receipt.

8.1.2 Fuel Handling, Transfer, and Core Verification

The procedures that provide administrative control of fuel transfer were reviewed. These included:

- SA-AP.ZZ-049 "CONDUCT OF FUEL HANDLING AND CORE ALTERATIONS
- RE-FR.ZZ-001 "FUEL HANDLING CONTROLS"

These procedures provide adequate control of the fuel transfers and the core verification.

8.1.3 Handling and Inspection of Other Core Internals

The procedures for the removal of various other core internals were reviewed. These included:

- MD-FR.KE-006 "REMOVAL AND INSTALLATION OF REACTOR VESSEL STEAM SEPARATOR AND STEAM DRYER"
- RE-FR.ZZ-002 "CONTROL ROD REMOVAL AND INSTALLATION"
- RE-FR.ZZ-003 "LPRM REMOVAL AND INSTALLATION"

These procedures provide adequate guidance and controls necessary to perform the tasks.

No violations were identified.

8.2 Administrative Controls of Plant Conditions

The following were reviewed:

- Hope Creek Technical Specifications
- OP-IO.ZZ-005 "Cold Shutdown to Refueling"
- OP-IO.ZZ-009 "Refueling Operations"
- OP-SO.EC-001 "Fuel pool Cooling and Cleanup System Operations"
- OP-SO.BC-001 "Residual Heat Removal System Operation"
- OP-AB.ZZ-101 "Irradiated Fuel Damage While Refueling"
- OP-AB.ZZ-144 "Loss of Fuel Pool Inventory"

The procedures needed to provide controls to place the mode switch in the refuel position are in place. They provide administrative control in the form of procedure sign-off's and hold points to ensure all the required surveillance testing is complete and that all Technical Specification conditions are met prior to entering the refuel mode. This includes demonstration of shutdown margin, ensuring adequate water level in the spent fuel pool and proper number of source range nuclear instruments are available. The procedures also provide the controls to handle work stoppage or delays.

Procedures are in place to operate the fuel pool cooling and cleanup system(FPCCS). The procedure for operating the Residual Heat Removal(RHR) system provides means to use the RHR system to augment to FPCCS during times of large heat loads in the spent fuel pool. This includes a full core off load immediately following a plant shutdown from an extended full power run.

There are abnormal operating procedures that address irradiated fuel damage, and the loss of fuel pool inventory. However, at this time there is no procedure that addresses loss of fuel pool cooling . This was brought to the attention of the operations manager. The loss of fuel pool inventory procedure will be revised in the future to include the loss of fuel pool cooling. This revision will be reviewed during a subsequent inspection of this area. During this refueling the loss of fuel pool cooling will not be a concern due to the off load consists of a third of the core. This heat load is small enough to be controlled by ambient losses.

No violations were identified.

8.3 Observation of New Fuel Receipt

The receipt and inspection of the first new fuel shipment was observed. This was done per W/O - 870819051 - "Receipt of new fuel". The reactor engineering department coordinated and tracked the progress of the work. The Hope Creek maintenance department unloaded the fuel, transferred the fuel to the refuel floor, and conducted an inspection of the fuel assemblies and channels. Adequate control of containment parameters was performed by the operations department. Radiation protection personnel were on scene to provide adequate control of radiological concerns. All efforts were coordinated in accordance with SA-AP.ZZ-039 "Receipt of New Fuel"

The new fuel inspection consisted of inspections of the upper and lower tie plates, rod to rod spacings checks, spacer inspections, simulated channel check, and overall inspection of fuel assembly cleanliness. These inspections were performed and the results documented in accordance with RE-FR.ZZ-004 "New Fuel Inspections"

During the fuel receipt and inspection there was adequate Quality Assurance personnel on the refuel floor providing surveillance. These individuals received the same training as the maintenance personnel performing the inspections.

After the inspections of the fuel assembly and channel were complete, the fuel was checked for proper rod to rod clearance on the inspection stand and then stored in the spent fuel pool. The spent fuel pool was dry at the time of the fuel movements allowing use of the polar crane to move the fuel.

All fuel movements were documented by the use of fuel movement sheets in accordance with RE-FR.ZZ-001. The fuel status boards were updated by the Reactor Engineering group.

No violations were identified.

8.4 Observation of Refueling Surveillance Test

The following surveillance test was observed:

- W/O 870423068
- MD-PM.DE-003 "Operational Test of Refuel Bridge"

This surveillance test was performed to prepare the refuel bridge and its associated cranes for the upcoming refueling outage. The work order also included a lubrication preventive maintenance which was completed prior to the operational checks. The operational check tested all limit switch settings, and performed load tests on the auxiliary cranes.

No discrepancies were noted.

8.5 Observation of New Fuel Inspection Training

The new fuel inspection training consisted of a classroom presentation and hands on training. The aim of the training was to qualify those personnel performing the inspections as well as those performing audits of the inspections as level I inspectors in accordance with ANSI N45.2.6-1978 "Qualification of Inspection, Examination and Testing Personnel for Nuclear Power Plants". The classroom portion demonstrated the information and techniques needed to perform and document the new fuel inspections in accordance with RE-FR.ZZ-004 "New Fuel Inspection".

The hands on training required that the trainee demonstrate two new fuel inspections to the instructor using a dummy fuel bundle.

The instructor was a qualified level III inspector from the Nuclear Fuels Division of General Electric contracted to conduct the training.

The inspector concluded that preparations for the refueling outage and the new fuel receipt inspections were thorough and well controlled.

9. Licensee Event Report Followup

The licensee submitted the following event reports during the inspection period. These event reports and periodical reports were reviewed for accuracy and timely submission. The asterisked reports received additional followup by the inspector for corrective action implementation.

Monthly Operating Report for October, 1987

* Special Report 87-007-00	Non-Compliance With FSAR Requirements for Meeting 10CFR50 Appendix R Criteria Due to Inadequate Review of FSAR Amendment 15
LER 87-040-00	Reactor Water Cleanup System (RWCS) Isolations (2) Following RWCS Pump Maintenance Due to Not Adhering to Procedures
LER 87-041-00	Overdue Channel Functional Test Due to Personnel Being Unfamiliar With New Computerized Scheduling System
* LER 87-042-00	Invalid Loss of Coolant Accident Signal Isolation When Performing Test Due to Leaking Instrument Valve
LER 87-043-00	Residual Heat Removal System Isolation While Performing Surveillance Test Due to Insufficient Work Space

LER 87-044-00 Residual Heat Removal System Isolation While Performing Surveillance Test

LER 87-045-00 Primary Containment Isolation System Initiation When Restoring Power to Logic Cabinet Due to Spurious Logic Module Inputs

* LER 87-046-00 Primary Containment Isolation System Initiation When Swapping Reactor Protection System Bus Power Due to Lack of Indication

* LER 87-047-00 Safety/Relief Valve Failure to Close - Sand Blasting Grit in Solenoid

9.1 Special Report 87-007 describes the electrical configuration of two reactor water cleanup (RWCU) system valves which were not in compliance with 10CFR50 Appendix R or Final Safety Analysis Report (FSAR) Appendix 9A. The two RWCU valves function as high to low pressure interface valves, and are identified in the FSAR as not having the required Appendix R, Section III.G.2 separation of electrical cabling. Appendix R requires cable separation to provide adequate assurance that a fire induced failure of one cable will leave the redundant train free of fire impairment. Hope Creek had committed in Appendix 9A of FSAR Amendment 15 to shut and remove power from the subject valves during normal operation. This action ensures that postulated worst case fire damage will not create a pathway for primary system leakage. The licensee found two RWCU valves closed but not de-energized. Corrective actions included revising procedures to include removing power to the subject valves prior to operational condition 1 and initiating a review of all FSAR amendments since Amendment 9 to ensure all FSAR commitments have been addressed. If either of these valves had failed open, no primary leakage would have occurred as the upstream isolation valve (HVF033) is normally shut. If HVF033 was open, primary pressure is reduced by two in series pressure reducers which ensure the downstream low pressure piping is not overpressurized. Any primary flow through the low pressure piping is directed to systems which collect and monitor water level, and alarm if a high level condition exists. The failure to de-energize these valves in the shut position was a licensee identified deviation from an FSAR commitment, and had minor safety significance. (NVD 354/87-23-02)

9.2 LER 87-42 describes an invalid channel "A" loss of coolant accident (LOCA) signal which was generated during a surveillance test. The surveillance test involved the calibration of a reactor vessel reference leg pressure transmitter. While pressurizing the isolated transmitter from a test source, the associated instrument isolation valve leaked, resulting in a pressure spike on the reference leg. The pressure spike actuated the level transmitter which produced the channel "A" LOCA signal. All systems responded to the LOCA signal as designed. The root cause of this incident was determined to be the leaking instrument isolation valve which was not seating properly.

Corrective actions consisted of repairing the leaking valve and re-performing the subject surveillance test.

9.3 LER 87-46 describes an actuation of primary containment isolation system (PCIS) channel "B" while swapping "B" reactor protection system (RPS) from the normal to alternate bus power. A PCIS isolation occurs upon receipt of two PCIS trip signals. Prior to the RPS power supply swap an undetected PCIS channel "B" trip had been received and not reset. When the "B" RPS bus was momentarily de-energized during the swap to alternate bus power, the PCIS channel "B" logic was satisfied and the isolation occurred. A contributing factor to this incident was the lack of indication in the control room of the existing PCIS logic status. LERs 86-006 and 86-055 document two additional PCIS isolations which are similar in that prior to PCIS isolation, an earlier, undetected PCIS trip signal was received and not reset. Corrective actions from these earlier LERs made instruction modifications to reset PCIS logic prior to performance and at the completion of all I&C procedures which produce a PCIS trip signal. The recent PCIS isolation involved an operations department procedure which had not been previously identified as producing a PCIS trip signal and therefore, had not been modified as part of the earlier corrective action.

Corrective action for this incident included revising the subject operations department procedure to reset PCIS logic prior to swapping RPS bus power supplies. In addition, the licensee will complete design change package 4-EC-1087 prior to the end of the first refueling outage to provide control room indication of PCIS logic status.

9.4 LER 87-047 details the failure of the "J" safety relief valve (SRV) to shut on command during plant testing. The plant was at 10% power when the "J" SRV was opened to collect baseline data for the associated acoustic monitor. When the SRV failed to respond to a close command, the reactor was manually scrammed in accordance with the abnormal operating procedure. The root cause of this event was determined to be sand blasting grit in the SRV solenoid valve. This is the first SRV failure attributable to sand blasting grit. Prior to initial fuel loading, sand blasting was used to clean damaged areas of the primary containment inner surface prior to touch up resurfacing. Vacuum particle capture techniques were used to minimize the spread of sand blasting grit. The licensee implemented the following corrective actions:

1. The malfunctioning SRV was replaced from stores, retested and determined to be functional. The remaining SRVs were visually inspected for damage.

2. Two sets of solenoid valves near "J" SRV and a random set from a SRV on the opposite header were disassembled and inspected for signs of sand blasting grit intrusion. No evidence of such intrusion was found.

   Additional details on the reactor scram are discussed in paragraph 3.2 of NRC Inspection Report 50-354/87-22.

No violations were identified.

10. Exit Interview

The inspectors met with Mr. S. LaBruna and other licensee personnel periodically and at the end of the inspection report to summarize the scope and findings of their inspection activities.

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