U. S. NUCLEAR REGULATORY COMMISSION

REGION I

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Report No. 50-354/87-29

Docket 50-354

License NPF-57

Licensee: Public Service Electric and Gas Company

Facility: Hope Creek Generating Station

Conducted: December 1, 1987 - January 4, 1988

Inspectors: R. W. Borchardt, Senior Resident Inspector
D. K. Allsopp, Resident Inspector
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Approved: P. P. Swetland 1/26/88
P. Swetland, Chief, Projects Section 2B Date

Inspection Summary:
Inspection on December 1, 1987 - January 4, 1988 (Inspection Report Number 50-354/87-29)

Areas Inspected: Routine onsite resident inspection of the following areas: operational safety verification, surveillance testing, maintenance activities, engineered safety feature system walkdown, assurance of quality, and licensee event report followup. This inspection involved 180 hours by the inspectors.

Results: A violation of station procedures relating to inadequate restoration of equipment after surveillance testing is cited in this report (paragraph 3). This violation is similar to procedural compliance problems identified in inspection report 50-354/87-23 and 50-354/87-24. Another licensee identified violation (not cited) involving an operations department surveillance test missed due to personnel error is discussed in paragraph 7. Several minor discrepancies in the fire protection area were highlighted in paragraph 2 for further licensee action and NRC review.

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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. Operational Safety Verification

2.1 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 compliance with 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 December 20, 1987 and January 3, 1988.

2.2 Inspection Findings and Significant Plant Events

The unit entered this report period at 100% power. A trip occurred on December 8, 1987 as discussed below. Otherwise, the unit remained at maximum allowable power levels throughout this report period except for short power reductions in order to perform maintenance or surveillance activities.

On December 4, 1987, the "B" Filtration Recirculation and Ventilation System (FRVS) recirculation fan flow transmitter was found to be isolated after the fan could not be started and kept running. I&C technicians had performed a calibration of the flow transmitter (GU-FT-0377B) earlier in the day and had failed to properly restore the instrument to service. This event was caused by a lack of strict procedural compliance on the part of the I&C technicians. Procedure IC-DC.ZZ-175(Q), "Tavis Pressure Transducer, Model P-8C Device Calibration" gives specific instructions for the return to service of the transmitter and also requires independent verification. However, these steps were not adequately performed. The licensee was informed that the failure to properly return the "B" FRVS fan flow transmitter to service constituted an apparent violation of station procedures (50-354/87-29-01). Although this violation was licensee identified it is being cited as a violation due to the recurring problems in the procedure compliance area. (Reference: NRC Inspections 50-354/87-23 and 87-24) The licensee has initiated a task force to resolve this and other procedure compliance problems.

At 2:04 p.m. on December 8, 1987, the reactor scrammed from 100% power. The scram was caused by a reactor protection system (RPS) channel "B" 1/2 scram signal generated by surveillance testing of the F APRM combined with an RPS channel "A" 1/2 scram signal caused by a spurious spike of the C main steam line (MSL) radiation monitor. At the time of the scram, I&C technicians were beginning to perform a surveillance test on the D MSL radiation monitor drawer when they experienced difficulty in pulling the drawer out of the cabinet. As they freed the drawer, the scram occurred. The post scram investigation concluded that the drawer was momentarily difficult to move because its top cover plate had worked loose and come in contact with the cabinet frame. The cause of the scram, however, was a frayed signal connector cable that allowed a high voltage signal (240 VAC) to momentarily short to the cabinet ground. Because the channel "C" and "D" MSL radiation monitoring cabinets have connected ground bus bars, the momentary short induced a spike in the C MSL radiation monitor. This spike was high enough to trip the monitor, and with the F ARPM 1/2 scram signal completed the RPS scram logic. The licensee had been aware for sometime that the C MSL radiation monitor trip setpoint was set ultra- conservatively low compared to the technical specification (TS) valve of three times normal. In fact, a design change package (DCP) had been approved days earlier to raise the setpoint to be consistent with TSs. Had there not been a delay between recognizing that the trip setpoint was low and completing the DCP, this scram may have been avoided. As part of the post scram review, the licensee held discussions with the General Electric Company who supplied the NUMAC MSL radiation monitor drawers to Hope Creek. The licensee inspected the other radiation monitor signal cables for fraying. No further discrepancies were found. The licensee's in-field inspections and the discussions with GE noted that the cabinet grounding configuration while not incorrect, may not be the optimum design. The current configuration did not cause the scram signal, however this area will be reviewed by the licensee to assess feasible improvements and will be reviewed by the resident inspector during the next report period.

The inspector witnessed the scram recovery from the control room and found all activities to be well controlled. The plant transient was mild. No emergency core cooling systems were required to actuate and no safety relief valves lifted. All safety related plant systems responded to the transient as designed.

The reactor was taken critical at 8:12 p.m. on December 9, 1987, after completing repairs to the main steam line radiation monitor signal cables. The main turbine was synchronized with the grid on December 10, and the unit returned to 100% power.

On December 10, 1987, the reactor water cleanup (RWCU) system isolated while attempting to return the B filter/demineralizer (F/D) to service. The F/D was not completely filled with water prior to attempting its return to service and when the F/D isolation valves were opened, a high delta flow condition was created. Similar isolations have occurred in the past and a system modification is scheduled for completion during the February 1988 refueling outage. The modification will enable the F/D to be filled and vented in a more complete and controlled manner. The RWCU system was returned to service with no further difficulties.

During this inspection period numerous minor infractions of fire protection procedures were noted. These infractions included fire doors which occasionally fail to shut and latch, fire extinguishers not properly mounted, and fire extinguishers which were not inspected on a monthly basis. Although these infractions are minor in nature, the number of infractions was of concern to the NRC. The resident inspectors will increase their inspection effort in the fire protection area during the upcoming inspection period to determine if additional regulatory action in this area is appropriate.

3. Surveillance Testing

3.1 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:

- CH-DC.ZZ-002 Calibration of Hydrogen/Oxygen Analyzers
- O -ST.KJ-001 "A" Emergency Diesel Generator Operability Test
- IC-FT.SE-016 APRM "D" Functional Test

No violations were identified.

4. Maintenance Activities

4.1 Inspection Activity

During this inspection period the inspector c[illegible]ed selected maintenance activities on safety related equ[illegible] 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.

4.2 Inspection Findings

Portions of the following activities were observed by the inspector:

Work Order	Procedure	Description
871229248	MD-GP.ZZ-002	Bolt, torque, and sequence requirements on "D" SACS instrument root valve
	MD-GP.ZZ-022	Disassemble, repair, and assemble "D" SACS instrument root valve

No violations were identified.

5. Engineered Safety Feature (ESF) System Walkdown

5.1 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, to 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.

5.2 Inspection Findings

The station service water system (SSWS) was inspected and in-plant conditions were found to be acceptable. SSWS provides cooling water to the safety auxiliary cooling system (SACS) and the reactor auxiliary cooling system (RACS) during normal operation. During a loss of coolant accident and/or loss of power accident SSWS provides cooling water only to the SACS heat exchangers. SSWS is also used as a backup means of filling the spent fuel storage pool, SACS head tank, and residual heat removal system.

A number of minor material and housekeeping deficiencies were noted, however, none were observed that adversely affected the safety related function of the SSWS. The majority of these deficiencies had been previously identified by the licensee and were being tracked for correction. In addition to in-field verification of system operability, the inspector verified that certain Technical Specification surveillance test requirements are satisfied through a review of the following procedures:

- OP-ST.KJ-005 Integrated Emergency Diesel Generator Test
- OP-ST.EA-001 Service Water Flow Path Verification
- OP-ST.EA-002 Service Water System Functional Test

All procedures were found to adequately fulfill the testing requirements of Technical Specifications.

The inspector noted that both SSWS bay watertight doors were inoperable due to empty nitrogen flasks used to supply gas to inflate the door seal. The "A" and "C" SSWS bay door also had a cracked nitrogen supply line to the door seal. The lack of routine maintenance on the SSWS watertight doors was identified by offsite safety review in September, 1986. Routine maintenance on these doors has been scheduled and implemented. The door seal is required to be inflated to ensure flood protection within one hour of notification of severe storm warnings. Restoration of the watertight door operability will be tracked as an inspector open item. (50-354/87-29-02)

No violations were identified.

6. Assurance of Quality-Control Room Environment

In addition to the normal routine observation of control room activities, a special assessment of the control room environment was conducted in accordance with Region I temporary instruction RI-87-01.

Control room activities have continued to be conducted in a highly professional manner. The performance of licensed operators and supervisors has been noteworthy, including their knowledge of plant status details. Significant progress has been made in reducing the number of overhead annunciators in alarm and this in turn has enabled the control room operators to focus their attention on more important plant parameters. The status of overhead annunciators as well as the general control room environment receives frequent attention from plant management (General Manager, Operations Manager,..etc.) and members of oversight groups (site quality assurance and onsite safety review). The presence of these individuals has promoted a continuation of the professional attitude developed during power ascension testing. The day-shift supervisor position is a one year temporary assignment for a senior shift supervisor who is used as an assistant to the Operations Manager. This individual spends a considerable amount of time in the control room monitoring plant and crew performance. The installation of carpeting in the control room has reduced the background noise level and personnel access control remains adequate.

The inspector identified a weakness relating to control room operator trainee's that was promptly corrected. The inspector noticed that licensed operator trainee's were not being required to conform to the same appearance, performance, and attitude standards as the on-shift licensed operators. The operations manager agreed with this observation and promptly took action to upgrade the trainee's control room demeanor.

Thorough shift briefings, the use of a work completion center outside of the control room, and the "whole plant" authority and responsibility given to the senior shift supervisor all contribute toward a professional control room environment.

7. Licensee Event Report Followup

The licensee submitted the following event reports during the inspection period. These event reports and periodic reports were reviewed for accuracy and timely submission. The asterisked reports received additional followup by the inspector for corrective action implementation. The (#) items identify reports which involve licensee identified Technical Specification violations which are not being cited based upon meeting the criteria of 10 CFR 2 Appendix C.

Monthly Operating Report for November, 1987

	LER 87-048	Trip of "B" and "D" Safety Auxiliaries Cooling System Pumps and Auto Start of "A" SACS Pump Due to Procedural and Design Deficiencies
*	LER 87-049	Primary Containment Leak Rate Determined in Excess of Allowable Leakage During Local Leak Rate Test (LLRT) Due to Component Malfunction
*#	LER 87-050	Missed Surveillance of a MSIV Outboard Steam Sealing Gas Test Line Isolation Valve

LER 87-049 describes 10CFR50 Appendix J Type "C" local leak rate testing (LLRT) which exceeded Technical Specification leak rate criteria due to a failed primary penetration. Technical Specifications require an overall integrated leak rate from all primary containment penetrations and valves to be less than or equal to 127,992 SCCM (La). During a LLRT conducted on April 9, 1987, a primary penetration (P-22) was determined to have failed when its individual leakage was calculated to be in excess of 100,000 SCCM. When this leakage was combined with other previously identified leakage, the overall leakage exceeded La. The leaking penetration was repaired and retested. The station carried out all required actions but due to a lack of station procedural guidance, did not make a 10CFR50.72 or 10CFR50.73 report. On December 3, 1987 during a review of the inservice inspection (ISI) test results, both ISI management and the licensing and regulation department determined that the event was reportable. They concluded that since La constitutes a design basis, the plant was operating outside design bases when the test determined leakage was in excess of La, and that reporting was required. Station administrative procedures were modified to include the requirement for reporting LLRT results in excess of Technical Specification limits.

LER 87-050 describes an overdue operations surveillance test on the outboard main steam isolation valve (MSIV) sealing gas test line isolation valve. On December 4, 1987 it was determined that this surveillance test, which was due on November 11, 1987, had been missed. Therefore, the outboard MSIV steam sealing system was potentially inoperable from November 11, 1987 to December 4, 1987. The isolation valve was tested satisfactorily and the outboard MSIV steam sealing system was declared operable. The operations department coordinator, who performs a second check on tracking the status of overdue operations surveillance test did not detect the missed surveillance test. This violation is not being cited as a violation as it was licensee identified and is the first operations surveillance test missed due to personnel error. (NV4 354/87-29-03)

8. 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.