January 10, 2005

Mr. A. Christopher Bakken, III
President and Chief Nuclear Officer
PSEG Nuclear LLC - N09
P. O. Box 236
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK NUCLEAR GENERATING STATION - PRELIMINARY RESULTS OF THE SPECIAL INSPECTION FOR THE OCTOBER 10, 2004 EVENT

Dear Mr. Bakken:

During the period of October 14 through December 16, 2004, the U.S. Nuclear Regulatory Commission (NRC) conducted a special inspection at the Hope Creek Nuclear Generating Station in accordance with Inspection Procedure 93812, "Special Inspection." This special inspection was conducted to assess the circumstances surrounding an event that occurred on October 10, 2004. Specifically, the plant was manually shutdown due to the failure of an 8-inch diameter moisture separator drain line, which discharges to the main condenser. By letter dated October 17, 2004, you provided the NRC with an overview of your plans to respond to this event. The NRC acknowledged your correspondence by letter dated October 21, 2004, from Samuel J. Collins, Region I Administrator. In that letter, the NRC stated that due to the heightened stakeholder interest in the event and consistent with NRC's openness strategic goal, the NRC would publish the preliminary results of the special inspection and meet with the public to review your actions and NRC findings prior to start-up of the Hope Creek facility.

The enclosure to this letter provides a summary of the inspection scope and preliminary inspection results in the areas reviewed. Please note that the final inspection results, including the number of findings and characterization of their significance, may change based on additional information and further review. The final inspection results, including any associated regulatory compliance issues, will be documented in NRC Inspection Report 05000354/2004013 which will be issued within 45 days after the inspection exit meeting scheduled for January 12, 2005.

The inspection focused on Hope Creek's investigation and root cause evaluations, including issue identification, extent of condition, potential common cause failures, root causes and corrective actions. The team independently evaluated the equipment and human performance issues that complicated the response to the event and assessed compliance with technical specifications and the emergency plan. Team members also evaluated the radiological releases associated with the event.

The team determined that PSEG's root cause evaluations were comprehensive and appropriately considered potential causes, extent of condition, and the problems encountered during the event. The inspection team confirmed that the root cause of the event was that personnel did not properly evaluate and address a degraded level control valve for the moisture separator drain tank. The level control valve malfunctioned several weeks prior to the event and caused the moisture separator drain system to operate in a condition outside its design. As a result, an 8-inch pipe in that system failed and caused the event on October 10, 2004. The assessment of this finding remains under review, but preliminarily, the finding is of low to moderate safety significance because it resulted in an actual plant event that included the loss of the normal power conversion system (the main condenser).

Overall, the team found that operator response to the transient was acceptable; however the operators were challenged by some equipment issues during the response to the event. Although these equipment problems challenged the operators, none of the problems would have prevented the systems from performing their intended safety functions or rendered the systems inoperable. The NRC inspection team identified three findings of very low safety significance associated with equipment and operational issues.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its Enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS).

Sincerely,

/RA/

Wayne D. Lanning, Director
Division of Reactor Safety

Enclosure: Summary of Inspection Scope and Preliminary Results

Docket No. 50-354
License No. NPF-57

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Inspection Scope and Preliminary Results

A. Inspection Scope

During the period from October 14 through December 16, 2004, the NRC conducted a special team inspection in accordance with NRC Inspection Procedure 93812, "Special Inspection," at the Hope Creek Nuclear Generating Station. The special inspection was conducted to assess the circumstances surrounding an event that occurred on October 10, 2004, involving the failure of an 8-inch moisture separator drain line, which discharges to the main condenser. The special inspection was initiated in accordance with NRC Management Directive 8.3, "NRC Incident Investigation Program," based on deterministic criteria and an initial risk assessment. Specifically, the condition involved possible generic implications, and involved questions pertaining to licensee operational performance. The initial risk assessment for this event was in the range where a special inspection was warranted.

The special inspection team consisted of five full-time members with expertise in the areas of plant operations, materials and mechanical engineering, and corrective actions. There were also four part-time members with expertise in the areas of emergency preparedness, radiological controls and protection, materials engineering, and probabilistic risk assessment.

The special inspection team was tasked to evaluate PSEG's analysis of the cause of the moisture drain pipe failure, extent of condition and actions to prevent recurrence, as well as to determine whether prior opportunities were available to prevent the event. The team was also tasked to develop an event chronology and independently evaluate human and equipment performance issues that complicated the response to the event. The inspectors also reviewed compliance with procedures and verified radiological releases were within regulatory requirements.

B. Preliminary Inspection Results

1. Plant Response: Personnel and Equipment Performance

The team reviewed and assessed licensed operator performance during the transient initiated by the moisture separator drain line failure until the plant was placed in the cold shutdown condition. The team provided particular focus on equipment issues that challenged the operators during the event. The team performed a detailed review of the data related to the event to assess overall equipment and human performance.

Results:

The team found that, overall, the operator response to the event was acceptable. However, there were some equipment issues that challenged the operators during the event and associated recovery.

Three findings that have been preliminarily determined to be of very low safety significance and one minor operator performance issue are described below.

High Pressure Coolant Injection (HPCI) System Valve Malfunction

PSEG determined that a limit switch had been incorrectly set for one of the two closed valves needed to satisfy an interlock to allow the HPCI full flow test valve (F008) to open. This self-revealing problem caused a delay of about five minutes when the operators attempted to place the HPCI system in service to control reactor pressure. The operators were able to satisfy the interlock and open the F008 valve by sending an additional close signal to the closed valve. The finding was considered to be of very low significance because it did not impact the accident mode of operation for the HPCI system, reactor pressure remained relatively stable during the period of time when HPCI operation was delayed, and alternate pressure control methods were available.

Reactor Core Isolation Cooling (RCIC) System Flow Oscillations

PSEG determined that operating experience regarding low flow limitations while operating the RCIC system in automatic flow control had not been incorporated into system operating procedures and operator training. As a result, the RCIC system was operated in a low flow condition (about 350 gpm) while in the automatic flow control mode and experienced unexpected oscillations. The RCIC system is normally aligned to operate in the automatic flow control mode in a high flow condition. During the event, the RCIC system had to be secured for approximately 10 minutes until the control system was adjusted. This problem was determined to be of very low significance since the RCIC system remained capable of performing its required safety functions, reactor water level was always maintained at least ten feet above the top of the active fuel, and the HPCI system was available for reactor vessel level makeup.

HPCI System Vacuum Pump Trip

PSEG determined that the wrong lubricant had been applied to the HPCI vacuum pump shaft. As a result, the HPCI system barometric condenser vacuum pump tripped several times during the depressurization and cooldown phase of the event. While the vacuum pump problem did not render the HPCI system incapable of performing its safety function, the operators decided to remove the HPCI system from service to prevent the release of radioactive effluents into the HPCI room due to operation without the vacuum pump. The finding was determined to be of very low significance because the HPCI system remained operable to perform its safety function without the vacuum pump.

Technical Specification Action Statement Interpretation

The team identified that the operators misinterpreted the Technical Specification Action Statement time requirement to place the plant into a cold shutdown condition within 24 hours of declaring the residual heat removal (RHR) system inoperable. During the event the operators aligned the RHR system to cool the suppression pool and declared the system inoperable in accordance with plant operating procedures. However, the operators misinterpreted the Technical Specification Action Statement time requirement and believed that they had 36 hours to complete the plant cooldown. The team believes this issue was of minor significance since the plant cooldown was completed safely and the RHR system could have been realigned to provide reactor makeup if needed.

2. Moisture Separator Drain Tank Piping Failure

The team reviewed the moisture separator drain tank system to determine how its design and operation may have contributed to this event. The team also reviewed the history of design and operational challenges associated with the system to determine if there were prior opportunities to identify, evaluate and prevent the conditions that led to the event.

Results:

The team found that engineers did not properly evaluate and recommend appropriate actions for a moisture separator drain tank level control valve problem. The level control valve problem resulted in a degraded condition that was outside of the system design basis and led to the steam pipe failure. The significance of this finding remains under review in accordance with the NRC's process for evaluating the significance of inspection findings.

Inadequate Evaluation and Corrective Action for Degraded Condition

PSEG determined that engineers did not identify that operation of the plant with the 'A' moisture separator drain tank level control valve (LV-1039A) failed open was outside the system design basis. Specifically, valve LV-1039A failed open on September 16, 2004, however, engineers did not recognize that continued operation in this condition placed the moisture separator drain system in a condition beyond its design capability. The open valve allowed the moisture separator drain tank to drain down which resulted in two-phase flow (a mixture of steam and water) through the moisture separator drain line to the main condenser. The two-phase flow introduced dynamic loading effects that had not been considered as part of the original design basis. This high dynamic loading caused the 8-inch moisture separator dump line to fail on October 10, 2004. In addition, engineers did not recognize, evaluate and properly address the fact that a similar condition occurred in 1988 and led to a crack in the same line.

The final disposition of this finding remains under review, however, preliminarily, it appears to be of low to moderate safety significance because the failure to correct the degraded condition resulted in an actual plant event that included the loss of the normal power conversion system (the main condenser). The main condenser was manually isolated by operators to terminate the steam break and required operators to use alternate means to depressurize and cooldown the plant.

3. Root Cause and Corrective Actions

The team evaluated PSEG's formal root cause evaluations associated with this event, including efforts to identify the cause of the pipe rupture, extent of condition reviews, and actions to prevent recurrence. The team independently evaluated personnel actions and equipment performance to assess the adequacy of PSEG's investigation.

Results:

The team determined that PSEG's root cause evaluations were comprehensive and appropriately considered potential causes and extent of condition for the pipe failure and the problems encountered during the event. The team determined that PSEG's proposed corrective actions were appropriate to address the identified problems and confirmed that corrective actions necessary for restart were implemented. The corrective actions included, in-part: a revised engineering decision making process, field walkdowns and inspections of pipe hangers and components, and revised operating and maintenance instructions to address the equipment problems which challenged operators during the event. There were no findings identified in this area.

4. Radiological Assessment

The team reviewed data and calculations used to quantify the amount of radioactive material released as a result of this event.

Results:

There was a small radiation release from the plant as a result of this event that was well below federally approved operating limits. Specifically, the total radiological release rate was less than 2% of Technical Specification limits. The total amount released was approximately 9.2 Curies of noble gas and consisted of both monitored and unmonitored release paths. A typical release for the same time period during normal operation would have been about 4.9 Curies. The unmonitored release occurred during a relatively short time frame (approximately 50 minutes) when steam was released to and exited the turbine building without transiting through the monitored ventilation exhaust path.

The team concluded that the radiological consequences of this event were negligible, and there were no findings identified in this area.