U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-354/95-16

License No. NPF-57

Licensee: Public Service Electric and Gas Company
P.O. Box 236
Hancocks Bridge, New Jersey 08038

Facilities: Hope Creek Nuclear Generating Station

Dates: August 12, 1995 - September 23, 1995

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Approved: Larry E. Nicholson, Chief 10/20/95
Projects Branch 3 Date

Inspection Summary:

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

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EXECUTIVE SUMMARY

Hope Creek Inspection Report 50-354/95-16

August 12, 1995 - September 23, 1995

OPERATIONS

Safe, uneventful operation was maintained throughout the inspection period. The inspectors noted improved communications in the control room, both orally and in log keeping. The number of alarm conditions in the control room were noted as increasing, which could lead to operator distraction; however, operators were observed to perform well in response to a leak in a steam jet air ejector. An engineered safety feature (ESF) actuation occurred involving an automatic swap of the high pressure coolant injection supply valves. This occurrence questions the effectiveness of prior corrective actions taken to prevent such actuations (see Section 2.1).

MAINTENANCE/SURVEILLANCE

Observed maintenance and surveillance activities continue to be performed safely and support plant operations. The corrective maintenance backlog was assessed as being appropriately managed, however, the backlog was getting larger and licensee identified performance goals were being exceeded. It was noted that the licensee's recently established Impact Plan (performance improvement plan) contains actions to address the increasing trend in outstanding corrective and preventative maintenance activities.

ENGINEERING

Station engineering activities generally supported plant operations well throughout the inspection period. Follow up to a repeat event involving a dropped new fuel bundle was thorough and resulted in a comprehensive root cause evaluation and subsequent corrective actions. Quality technical department monitoring and analysis of safety related system performance data was evidenced during an inspector review of elevated safety relief valve tailpipe temperatures and control room emergency filtration system heater surveillances. Support of planned and forced system maintenance outages continued to be a strength. However, persistent reliability problems with effluent release path radiation monitors indicated that engineering resolution of this issue has been less than effective. Communications with the NRC regarding licensee review of Generic Letter 95-07, pressure locking and thermal binding of safety-related motor-operated valves was considered excellent. NRC review of this issue subsequent to this report concluded that adequate basis existed to support valve operability.

EXECUTIVE SUMMARY (Continued)

PLANT SUPPORT

Plant support activities were observed to be very good. While some necessary hardware, mostly associated with radiation monitoring system and radioactive waste handling were degraded, radiation protection and chemistry personnel were performing appropriate tasks to ensure releases were monitored and quantified. The inspectors noted improved communication with station personnel regarding exposure data to better track activities causing personnel exposure.

SAFETY ASSESSMENT/QUALITY VERIFICATION

Very good self-assessment activities were observed during this period, including Quality Assurance, Safety Review Group and Station Operations Review Committee activities. The Nuclear Review Board was assessed as providing excellent critical review of ongoing station activities.

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DETAILS

1.0 SUMMARY OF OPERATIONS

The Hope Creek unit began the inspection period at 100% power and uneventful operations were maintained for the duration of the inspection period. At the end of the inspection period the unit had been maintained on-line continuously for 61 days.

2.0 OPERATIONS

2.1 Inspection Findings and Significant Plant Events

The inspectors verified that Public Service Electric and Gas (PSE&G) operated the facilities safely and in conformance with regulatory requirements. The inspectors evaluated PS&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 5 hours of deep back-shift inspections.

End-of-Cycle Recirculation Pump Trip System Instrumentation

The inspectors noted this report period that both channels of the end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation, required to be operable by Hope Creek technical specification 3.3.4.2, have been in "bypass" since August 1993. Operators appropriately entered the associated action statement which allows plant operation to continue provided that minimum critical power ratio (MCPR) is greater than or equal to the EOC-RPT inoperable limit specified in the core operating limits report. The inspectors determined that, though this action resulted in a reduced margin to the MCPR thermal limit, it was technically acceptable. When questioned why entry into this action statement was allowed to persist for over two years, operators stated that thermal stratification in the reactor vessel following a scram would be precluded by eliminating a recirculation pump trip, which would result in a net safety benefit.

In addition to the above noted long-term action statement entry, operators implemented a modification that extinguished the overhead annunciator normally activated when the EOC-RPT instrument is bypassed. Operators stated that this action was taken to reduce the overall number of active annunciators in the control room.

The inspectors independently verified that all necessary action requirements were implemented, and that the deactivated overhead annunciator was tracked in the degraded annunciator log ("DL-10" log). With the exception of monthly channel functional testing, all required technical specification surveillances were current for the EOC-RPT instrumentation. Though all the actions taken in this case were legally acceptable, the prolonged entry into technical specification action statements constitutes a poor precedence.

Primary Containment Isolation Signal Bypassed for Routine Maintenance

During the week of September 19, 1995, operators opened the breaker for the motorized actuator on the reactor water cleanup system inlet isolation valve, 1BGHV-F001, in order for maintenance technicians to perform a reactor protection system motor generator output voltage adjustment. Operators stated that this breaker was routinely opened during this adjustment in order to preclude the possibility of an inadvertent trip signal being generated which would isolate the reactor water cleanup system. In this particular example, the noted containment isolation valve was inoperable for approximately one hour.

The inspectors verified that the appropriate technical specification action statement for an inoperable containment isolation valve (3.6.3.a) was entered. However, despite the fact that technical specification requirements were met, the practice of bypassing the isolation function was not addressed in either the operations department system operating procedure (HC.OP-SO.SB-0001) or the maintenance department work procedure (HC.MD-CM.SB-0001). The inspectors concluded that, though regulatory requirements were satisfied in this case, bypassing the containment isolation function of the F001 valve to complete the work may not be appropriate from a risk perspective. This matter will be tracked as an inspector follow up item. **(IFI 50-354/95-16-02)**

Steam Jet Air Ejector Leak

On September 7, 1995, a steam leak from the inlet piping of the "B" steam jet air ejector. Operator response to this transient was good, and resulted in a successful transfer to the "A" air ejector with minimal impact on the plant. Effective shift response was particularly noteworthy in light of the fact that several equipment degraded conditions hindered operators during the event. Examples included a motor operated valve malfunction, an inadequately set steam supply pressure controller, and failed overhead annunciator. All of the noted problems were subsequently identified in a condition report and were being addressed at the conclusion of the report period.

The inspectors concluded that, in addition to the effective operator response to the initial transient, post-event corrective actions were good and maximized the integrity of the condenser vacuum boundaries and minimized offgas release flow. For example, when operation of the "B" air ejector was terminated, the steam leak became a vacuum leak which reduced condenser efficiency and increased offgas flow. Follow up radiography surveys of the air ejector steam piping identified other areas of pipe wall thinning which were subsequently repaired by encapsulation to temporarily restore the system for limited use.

Hope Creek Operations Department Performance Assessment

The inspectors reviewed the Hope Creek Operations Performance Assessment, dated August 30, 1995. This assessment was done at the request of the General Manager-Hope Creek Operations to determine if the Operations Department's performance since January 1994 had degraded, to identify weaknesses, and recommend long-term improvements. This assessment was committed to the NRC as part of the corrective actions for the July 8, 1995, shutdown cooling bypass event. The assessment utilized resources, technology and methodology developed by Failure Prevention Incorporated (FPI) International.

The assessment noted that Licensee Event Reports (LERs) have increased since 1993. A rolling quartile average of reportable events has doubled from about 3/QTR to 6/QTR. The assessment also noted an increased number of events caused by the operations department.

The assessment identified a number of concerns that contributed to the causes of the above increased event rate. Among these concerns are the following: lack of a mission resulting in poor work prioritization; poor vertical communications, resulting in a failure to communicate management expectations; a perceived poor accountability program; a decreasing trend in knowledge and experience for on-shift personnel; and a significant backlog in procedure revisions. On the positive side, the assessment also identified that the operations department generally works well with other organizations. The assessment made a number of recommendations to improve operations performance. The recommendations were clearly defined and address the identified concerns and were rank ordered according to the FPI key characteristics of a strong organization.

The inspectors concluded that the assessment of operations was an excellent tool to improve performance and provided critical analysis and insight regarding the causes of the increased arrival rate of operational events. The report independently assessed and clearly articulated a number of issues that have concerned the NRC throughout the last year.

Reportable Events

During this period Hope Creek notified the NRC pursuant to 10 CFR 50.72 of three events. The first event involved an automatic actuation of the high pressure coolant injection (HPCI) suction flow path, which is considered an engineered safety feature actuation, that occurred on September 8, 1995. The cause of this event was an apparent high level in the torus that automatically shifted HPCI suction flow path from the condensate storage tank to the torus.

This was the fourth such actuation of the HPCI suction flow path in the last two months. Two prior events were due to scheduled activities known to likely result in the actuation and were not reportable. However, the first similar event, that occurred on July 3, 1995, and was described in Licensee Event Report 50-354/95-014-00, was nearly identical to this most recent occurrence. Licensee management expressed concern that the corrective actions for that event did not prevent recurrence, and planned a more comprehensive analysis of the current event to include a review of the prior event's corrective action effectiveness. The inspectors consider this matter unresolved pending review of the applicable licensee event report. **(URI 50-354/95-16-01)**

The second event involved a potential loss of the safety parameter display system (SPDS) reported on September 19, 1995. The licensee found under certain postulated conditions due to a degraded power supply, that the SPDS would not be immediately available for loss of coolant accident/loss of offsite power (LOCA/LOOP) conditions. Subsequently, on October 6, 1995, this event report was retracted since the SPDS was in fact available at the time of the event; and, while operator action would have been required to restore the system to operation following a LOCA/LOOP, this did not mean the system was inoperable. The resident inspectors verified through interviews with on-shift personnel that: (1) they were "tracking" the SPDS while the normal power supply was degraded so that they knew that manual action would be required if a LOCA/LOOP condition occurred; and, (2) that the control room staff, including the shift technical advisor (STA), train on LOCA/LOOP response, including conditions such as a loss of SPDS. Also, the inspector found that the STA procedure for monitoring accident conditions does not require SPDS availability in order for the procedure to be completed. The inspector assessed that the initial report of this condition was very conservative by the control room staff and that based on additional review by the plant staff, the retraction of the report was appropriate.

The third event report involved HPCI being declared inoperable on September 20, 1995, due to unacceptable water concentration in the HPCI turbine oil reservoir. The specified maximum water content is 0.20% and a routine sample indicated 0.23%. Operator actions regarding this event were reviewed and found to be acceptable.

Subsequent to the report, the licensee determined that the most probable cause of the water intrusion was steam leakage past the HPCI turbine steam emission valve and stop valve. The steam condenses and mixes with the turbine bearing oil and subsequently collects in the oil reservoir. The oil in the reservoir was removed and replaced with new oil. A new routine of: sampling weekly; running the barometric condenser daily for approximately ½ hour; and, removing approximately 5 gallons of oil from the reservoir bottoms weekly has been established to continually remove moisture from the turbine and oil and verify that the water content does not exceed the specified limit.

The resident inspectors found that the immediate response by on-shift operations personnel and subsequent response by engineering were very good. The leaky valve in the steam supply cannot be corrected until the upcoming refueling outage; however, the interim measures developed should minimize the impact of the water intrusion as well as measure the effectiveness of those actions to ensure that the water content in the oil does not result in HPCI inoperability.

Overall, the inspector assessed that the control room personnel met the reporting requirements of 10 CFR 50.72 for the events described above.

Control Room Observations

Throughout the report period, the inspectors witnessed several improvements in routine plant operations. Examples included "three-way" communications among operators, both in person and over radio circuits. Reactor operator and shift supervisor log entries were made for lower significance issues and were more detailed and descriptive. Pre-job briefings also improved, with increased participation and questioning from attending personnel. The inspectors noted that the removal of loose cellophane insulation from a recirculation pump motor generator exciter housing was well controlled and coordinated. Enhancements were made in operations control of plant maintenance activities by instituting a full time work control staff.

The inspectors also observed an increasing trend in the number of inoperable control room instruments, alarms, and indicators. In addition, there were several occasions this report period when there were more than twenty overhead annunciators illuminated in the control room simultaneously.

Safety Auxiliaries Cooling System Pump Shifting to Manual

The inspectors noted that, on numerous occasions during the report period, the "C" safety auxiliaries cooling system pump automatically shifted from automatic to manual control. This problem was initially listed as an operator workaround in 1994. Hope Creek engineering personnel stated that the cause of this shift in pump control was due to spikes in the output of the associated loop flow transmitter, and that numerous attempts were made to resolve the problem. To date, however, none of these attempts have been successful in totally eliminating the problem.

The inspectors observed that operators normally returned the pump control to automatic immediately following an inadvertent shift to manual. On September 8, 1995, after several unplanned transfers to manual control, a reactor operator inadvertently secured the "C" pump during an attempt to place it back in automatic. Recognizing his error, the operator quickly restarted the pump and restored the safety auxiliaries cooling system to its normal configuration.

The inspectors concluded that, despite the appropriate characterization of this problem as an operator workaround, the inadvertent shifting of the "C" pump to manual was a distraction to control room operators that ultimately led to a personnel error. The inspectors learned that a design change was developed that would replace the noted flow transmitter output amplifier and resolve this persistent issue.

3.0 MAINTENANCE/SURVEILLANCE TESTING

3.1 Maintenance Inspection Activity

The inspectors observed selected surveillance and maintenance activities on safety-related and important-to-safety equipment to determine if PS&G conducted these activities in accordance with approved procedures, technical specifications, and appropriate industrial codes and standards. Routine observation of daily planning meetings and discussions relative to net positive safety gain for on-line maintenance activities were generally assessed as positive indicators. In general, the activities observed were judged effective in meeting the safety objectives of the Hope Creek maintenance and surveillance program, except where specifically noted otherwise.

3.2 Inspection Findings

Control of Corrective Maintenance Backlog

The inspectors reviewed the planning, scheduling, and monitoring of backlogged corrective maintenance at the Hope Creek station. As of September 18, 1995, the total backlog level was approximately 840 work orders, each of which had a variable number of individual work activities associated with them. By comparing this figure with station historical data, the inspectors determined the current backlog was above normal levels for the plant at this point in an operating cycle. The inspectors further noted that the current backlog was at the highest level it has been this operating cycle, partly because of a recent change in the way backlog was tracked and partly because some previously planned refuel outage work had been rescheduled for conduct while the plant was on line. The station's established goal for maximum corrective maintenance backlog was 525 work orders.

The inspectors assessed the prioritization scheme employed by the planning department to establish how a particular corrective maintenance activity was scheduled. All backlogged corrective work was tracked in an electronic database and monitored by its planning status (e.g. on hold for parts, ready to work, in planning, etc); however, the backlogged work was not further distinguished as being safety related or non-safety related. As a result, the inspectors could not assess the overall impact of backlogged safety related work orders. While printouts could be generated to list all backlogged work by system designator, an impact analysis to determine the net effect on safe plant operation would be a cumbersome prospect.

All backlogged work was provided a priority level for scheduling purposes, based primarily on the particular item's impact on personnel safety and/or plant system reliability. The inspectors noted that station work control procedures and planning standards have established time limits for each priority level, by which time specified work should be completed. However, in practice, except for the highest priority levels, these limits were frequently not adhered to. These priority levels were more commonly used in a relative sense for work scheduling as opposed to use as an absolute "time to work completion" standard. In addition, though tracking information was available to indicate the total number of corrective maintenance work orders by age (in months), data indicating the age of work orders by priority level (to assess adherence to time standards) was not readily available or commonly analyzed by the planning department.

Finally, in the case of planned and forced system outages, the inspectors reviewed how backlogged corrective maintenance was scheduled for inclusion. The inspectors concluded that, in these situations, corrective work was generally performed unless it was on hold (for parts, engineering, etc), of minimal significance, or would increase the overall outage duration by an unacceptable margin. However, for safety system outages, the inspectors determined that the station did not typically perform an assessment of the combined impact of corrective maintenance backlog on the required redundant systems. This assessment was made in a more global sense by operations personnel based on known degraded conditions of the redundant systems (i.e. review of so-called "tracking LCO's").

The inspectors concluded that corrective maintenance backlog at the Hope Creek station was adequately prioritized, scheduled and monitored. Available performance indicators were generally informative of the nature of the backlog in terms of total number, trend and age. However, weaknesses were noted in PSE&G's ability to assess the overall safety impact of this backlog, adhere to established time standards for work completion, and understand how effectively backlogged work in individual priority levels was being worked off. It was finally noted by the inspectors that part of the licensee's Impact Plan will address the increasing station maintenance backlog.

Maintenance and Surveillance Observations

The inspectors observed numerous maintenance and surveillance activities this report period, including close observation of the "D" emergency diesel generator extended outage, the reactor core isolation cooling system forced outage for jockey pump piping replacement, and the quarterly inservice test runs of both the high pressure coolant injection and reactor core isolation cooling systems. In all cases, pre-job briefings were of high quality, with good attendance from each department associated with the work. In addition, excellent supervisory oversight and quality assurance monitoring was observed. Where applicable, tagging boundaries were reviewed and considered appropriate. Work orders and procedures were immediately available at the job sites and were being appropriately adhered to. For work involving cutting, grinding, or welding, appropriate controls were established to ensure the potential for fire was minimized and post maintenance fire watches were stationed in accordance with administrative procedures. Post maintenance testing was appropriate to establish a basis for returning the affected systems to operable status.

Scram Discharge Volume Drain Valve Failure

On September 9, 1995, air-operated scram discharge volume drain valve 1BFHV-F011, one of two valves in series, failed to fully close during a monthly stroke test. The valve, designed to fail closed by spring force when air is vented off the air actuator (safety mode), did not achieve a fully shut position during the test because an adjustment collar on the valve stem rotated to a position that impeded full valve motion. Maintenance and technical department personnel subsequently determined that the adjustment collar was not positively secured in place by a locking device, which ultimately allowed the collar to move and inhibit proper stroke length. As a result, all similar valves were checked for indications of a similar problem and were found satisfactory. Adjustments were made to 1BFHV-F011 and the valve was satisfactory retested.

The inspectors concluded that Hope Creek's response to this issue, including problem identification, resolution and generic review was very good. The system engineer, assigned followup review for this potentially generic condition, determined that a design change (to add an adjustment collar locking device) was necessary to ensure that this failure mode would not repeat. Further, consideration for 10 CFR Part 21 reportability was planned since the valve and actuator manufacturer was relatively common in the nuclear industry (Hammel Dahl).

4.0 ENGINEERING

4.1 Inspection Findings

Dropped Fuel Bundle

On August 23, 1995, while seating a new GE9B fuel bundle into the new fuel inspection stand on the refuel floor, a casting nub on the bundle's upper tie plate caught on the fixed member of the inspection stand upper retaining ring. This caused the hoist cable (suspending the bundle) to momentarily slacken. Before technicians could react, the bundle freed itself from the obstruction and dropped approximately two inches into the base of the stand. Technicians completed the bundle inspection and did not identify any visual indication of damage. The bundle was ultimately shipped back to the vendor.

The inspectors noted that this event was similar to a previous dropped fuel event that occurred on January 7, 1994, and documented in NRC inspection report 50-354/93-27. In that case, PSE&G attributed the cause of the event to procedural weaknesses and inattention to detail. Corrective actions included procedural upgrades to improve coordination of fuel moves and counseling of the individuals involved.

In this recent occurrence, the cause of the event was more appropriately classified as a design deficiency in that the bundle tie plate protrusions could freely catch on the inspection stand upper retaining ring. The inspectors determined that subsequently developed corrective actions were clearly focused on eliminating this root cause. Specifically, "channel caps" (made by cutting a one foot section from the top unused fuel channels) were placed over the bundle upper tie plates prior to their transport to the inspection stand. In addition, horizontal stripes were marked on these caps to aid in positioning the bundles appropriately in the stand prior to ungrappling. These channel caps were thoroughly tested using a dummy bundle to ensure the problem was resolved.

New fuel inspection resumed on August 28, 1995, after bundle handling procedures were modified and appropriate training was completed. The inspectors observed subsequent fuel handling and concluded that implemented corrective actions were appropriate. The inspectors further judged that, despite the previous opportunity to identify appropriate root causes, this recent event was a case in which the actual root cause was clearly identified and effective actions were taken to preclude recurrence.

In addition to the above, the inspectors conducted a review of the potential generic implications stemming from a recent fuel handling event at the Susquehanna Steam Electric Station. In this case, the fuel handling grapple separated from the hoist cable which allowed suspended new fuel to drop in the spent fuel pool. The inspectors noted that Hope Creek refuel floor supervisors and reactor engineering personnel demonstrated good awareness of the Susquehanna event. Independent verification determined that the Hope Creek grapple was a fundamentally different design than the Susquehanna grapple, and that Hope Creek procedures required positive verification of grapple integrity prior to each fuel move. Implementation of this procedural requirement was observed in practice.

Control Room Emergency Filtration Heater Surveillance Testing

During a review of surveillance procedures, a Hope Creek system engineer determined that testing designed to demonstrate the operability of control room emergency filtration system heaters did not adequately measure dissipated kilowatts necessary to satisfy technical specification 4.7.2.e.4. Specifically, the bus voltage used to calculate power dissipated by the heaters was a nominal value rather than an actual measured value. The engineer recognized that surveillance test procedures for the filtration, recirculation, and ventilation system heaters were similarly flawed. A second concern was also identified by the engineer that if actual measured bus voltages exceeded specific thresholds (that were still within allowable technical specification tolerances), the heater power measurements would fail to meet satisfactory surveillance test criteria.

The inspectors reviewed the follow up actions performed in response to this issue and concluded that engineering personnel performed well in every aspect, from problem identification to resolution and generic review. Hope Creek engineering personnel revised the affected surveillance procedures to properly account for actual bus voltages, including a method to normalize these values to ensure test-to-test heater KW trend information would be valid. This new calculational methodology was "backfitted" into prior test data to ensure that previously conducted surveillances demonstrated satisfactory results.

An engineering review to resolve the second concern (noted above) concluded that, since the heater power surveillances were intended to be functional tests to trend heater performance, the concern regarding the ability of the heaters to satisfy surveillance criteria under all conditions of possible bus voltages was unsubstantiated. Finally, Hope Creek personnel contacted other utility representatives to determine whether the noted concerns were generic in nature.

Safety Relief Valve (SRV) Tailpipe Temperature Monitoring

Three of the fourteen two-stage Target Rock safety relief valves at Hope Creek continued to experience elevated tailpipe temperatures this report period, evidence of pilot and/or main seat leakage. The valves in question ("E", "K" and "L) indicated tailpipe temperatures between 210 and 215°F, very close to the 220°F alarm setpoint. These temperature indications were directly observable in the main control room on a recorder. Individual tailpipe acoustic monitors, a technical specification required redundant instrument, did not indicate any abnormal changes in valve positions. On two occasions, the "L" valve briefly exceeded the temperature alarm setpoint. Operators responded appropriately by checking redundant indications and engaging maintenance and technical support to evaluate the validity of the signal.

The inspectors reviewed safety relief valve trend data with the responsible Hope Creek system engineer. The engineer stated that, despite evidence of seat leakage on the noted three valves, tailpipe temperatures were not yet at a level that necessitated serious concern. He further indicated that two of the suspect valves ("E" and "K") have shown indication of leakage since the previous refueling outage, and the other began a slowly increasing temperature trend in June 1995. Based on consultations with the valve manufacturer, the engineer believed that the elevated temperatures were in a range that could indicate pilot seat leakage, possibly caused by foreign material left on the seat following full pressure testing.

The inspectors concluded that Hope Creek engineering personnel were effectively monitoring the condition of all fourteen safety relief valves. Further, engineering demonstrated a thorough knowledge of other indications of this leakage, including the frequency which torus letdown and cooling evolutions were conducted. Finally, industry event information related to Target Rock valves was actively pursued and evaluated for impact on Hope Creek. The engineering organization has been in contact with counterparts at the Limerick station and have incorporated lessons learned from the Limerick September 11, 1995, SRV lift into their review and assessment of SRV leakage.

Generic Letter 95-07, Pressure Locking and Thermal Binding of Safety-Related Motor-Operated Valves

The licensee informed the NRC resident and Region I Offices of some concerns regarding potential insufficient margins for eighteen motor-operated valves regarding susceptibility to pressure locking or thermal binding failures. The valves were all in emergency core cooling system applications. Eight such valves mostly associated with core spray had very small margins and were of most concern to both the licensee and NRC. The licensee provided additional calculations to the NRC regarding the valve motor output thrust and calculated closing/opening forces given the possible pressure locking and thermal binding consideration. The licensee considers the valves to be operable at this time based on their margin calculations. However, plans have been made to modify the valve internals during the next refueling outage to reduce the susceptibility of the valves to this type of failure mechanism. The NRC review of this issue, subsequent to this report period, concluded that adequate basis existed to support valve operability. This review and conclusion will be contained n a future inspection report to closeout unresolved item 50-354/94-24-01. The inspectors concluded that the licensee's handling of this issue and communication of relevant information to the NRC was excellent.

4.2 Followup of Prior Inspection Findings

(Closed) Unresolved Item (50-354/95-11-03) High Pressure Coolant Injection Torus Suction Valve

During the previous report period, the inspectors questioned why the high pressure coolant injection system torus suction valve (1BJHV-F042) was listed in technical specification Table 3.6.3-1.A.5(b) for automatic containment isolation valves. The concern centered on the need for the high pressure coolant injection system to remain functional during loss of coolant accident scenarios. This function would be nullified if the torus suction valve were to automatically close on a primary containment isolation signal. Though a review of the system's design and previous testing clearly demonstrated that the intended safety function would be satisfied, the inspectors could not determine if system design basis documentation accurately reflected regulatory requirements.

During this report period, Hope Creek engineering personnel met with inspectors to resolve the apparent discrepancy between the plant's technical specifications and design bases. The engineers verified that 1BJHV-F042 was in fact a remote manual valve, and would not automatically close on a primary containment isolation signal. The engineers presented applicable design basis documentation to demonstrate the valve's compliance with Standard Review Plan (NUREG 0800) requirements and 10 CFR 50 General Design Criteria. The inspectors concluded that, based on an independent review of the noted regulatory requirements, actual plant design documentation and system functional testing data that the high pressure coolant injection torus suction valve would perform its intended safety function and that the basis for its design and operation were adequately documented. This item is closed.

5.0 PLANT SUPPORT

5.1 Radiological Controls and Chemistry

The inspector periodically verified PS&G's conformance with their radiological protection program. During plant tours and direct observation of operations and maintenance activities, the inspector observed that the radiological protection program was being properly implemented.

Radioactive Effluent Pathway Monitoring

On several occasions during the report period, the inspectors observed that a large percentage of the Hope Creek effluent release paths had simultaneously inoperable radiation monitors. Most of the affected monitors, including those for the offgas, liquid radwaste and filtration recirculation ventilation systems have experienced chronic reliability problems. In addition, some of the radwaste processes were also not performing appropriately. The inspectors independently confirmed that all technical specification action requirements associated with the inoperable monitors were implemented within the allowable time restrictions. Further, all of the deficiencies were appropriately entered into the maintenance database for resolution. However, despite the fact that all operational requirements were met for effluent monitoring, the inspectors concluded that, based on the persistent nature of many of the identified problems, corrective actions to date were not successful in minimizing distractions to plant operators and reducing the resulting potential for increased personnel errors.

Radiological Control Area (RCA) Egress Process Change

On September 11, 1995, the radiation protection department noted that greater than 300 mRem exposure occurred that day. Daily exposure rates are normally between 50 and 150 mRem. After reviewing dose records for various jobs in the RCA, it was determined that most of the dose on September 11 occurred on minor dose rate jobs and for individuals on routine tours. The radiation protection personnel verified that there had not been a change in RCA radiological conditions that would account for the increased exposure. As a result, the RCA egress process has been changed so that any individual receiving greater than 10 mRem will document the nature of the work activities that they were involved with that day. The inspector found that the licensee's handling of this unusual exposure to be very good, in that, they first verified that no changing radiological conditions were a cause. It was also noted that since this occurrence, radiation protection now provides daily assessment information to all supervisory personnel at the daily Plan-of-the-Day meeting, increasing communications of radiation protection information to the station staff and raising awareness of personnel to radiological conditions.

5.2 Emergency Preparedness

The inspector reviewed PS&G's conformance with 10 CFR 50.47 regarding implementation of the emergency plan and procedures. In addition, the inspector reviewed licensee event notifications and reporting requirements per 10 CFR 50.72 and 73. During this inspection period there were no required emergency notifications.

5.3 Security

The NRC verified PS&G's conformance with the security program, including the adequacy of staffing, entry control, alarm stations, and physical boundaries. The inspectors observed good performance by Security Department personnel in their conduct of routine activities. During tours of the protected and vital areas, the inspectors observed that the security related hardware was maintained in good working order. The inspectors observed the implementation of actions taken relative to preventing unauthorized vehicle entry to the site. These activities appeared to be well controlled.

5.4 Housekeeping

The inspector reviewed PS&G's housekeeping conditions and cleanliness controls in accordance with nuclear department administrative procedures. During routine plant tours and in system restoration after maintenance activities, the inspector observed generally good implementation of the station cleanliness program.

5.5 Fire Protection

The inspector reviewed PS&G's fire protection program implementation in accordance with nuclear department administrative procedures. Items included fire watches, ignition sources, fire brigade manning, fire detection and suppression systems, and fire barriers and doors. The inspectors noted that the licensee identified and corrected minor deficiencies relative to combustible material storage containers within the plant.

6.0 SAFETY ASSESSMENT AND QUALITY VERIFICATION

Radioactive Waste Processing System

Over the course of the last several report periods, the inspectors witnessed an increased level of scrutiny of radioactive waste processing equipment and procedures by Hope Creek station personnel. The inspectors noted that recent comprehensive reviews by both the onsite safety review group and the station quality assurance department were particularly effective in identifying valid radioactive waste system concerns. For example, an issue raised regarding repeat failures of the liquid radwaste monitor (used to terminate offsite discharges on high activity) resulted in the initiation of a comprehensive engineering effort to understand root causes and identify corrective actions. The increased level of internal assessment also resulted in an engineering review to determine whether use of outside contractor radwaste processing technologies would prove more cost effective than continued reliance on existing plant equipment. The inspectors concluded that recent self assessment activities of the radioactive waste system was evident of an improved Hope Creek program for identifying and documenting plant problems.

Technical Specification Surveillance Improvement Project

In September 1995, PSE&G senior management chartered a full-time comprehensive review team to perform a 100% verification of the adequacy of all Hope Creek technical specification surveillance procedures. This team, led by a Hope Creek manager and staffed with approximately six contract engineers, was expected to complete its evaluation by late 1996. The inspectors concluded that, based on the number of recently identified discrepancies in this area, this project was an excellent initiative to understand the scope of the problem and resolve identified issues in a timely manner.

Nuclear Review Board (NRB) Meeting

The inspector observed portions of the NRB meeting held on August 29-30, 1995. The NRB is a non-technical specification, management self-assessment group that provides recommendations to the President, Nuclear Business Unit based on their independent assessment of station activities. The inspector observed presentations to the NRB relative to the new corrective actions program, the recent results of QA oversight activities, recent Hope Creek operations, and ongoing management review of the July 8, 1995, shutdown cooling bypass event. The inspector also reviewed a sampling of prior NRB meeting minutes and recommended actions for improvement. The inspector concluded that the NRB provides very useful, critical analysis of Hope Creek performance, and as a self-assessment activity is an excellent tool to avoid future problems and to find ways to improve operational performance.

Shutdown Cooling Special Review Team

The inspector periodically met with the special review team that was convened to assess prior licensee corrective actions regarding events involving actual or potential losses of shutdown cooling. This team had very good representation from various parts of the licensee organization and also had external participants, including INPO and GE. The results of this review, as well as any planned corrective action on the part of the licensee will be documented in a future report when the assessment is complete. The team exit is planned for October 6, 1995, and the report conclusions and recommendations should be available by the end of October.

Station Operations Review Committee (SORC) Review of the Reactor Core Isolation Cooling (RCIC) Battery Charger

The inspector observed a number of SORC activities during the period. In general, SORC requirements were met. As an example of a good SORC activity, the inspector observed a review of an operability determination prepared by the Nuclear Engineering Department (NED) for the degraded RCIC 250V battery charger. During August and September the RCIC battery charger began to experience momentary, periodic high voltage spiking conditions that would cause the charger to trip. Operators would then have to manually reset the trip and the charger would operate without any other notable problems. The operators considered the charger degraded, but operable and requested NED to provide an analysis supporting the operability determination. Normally these engineering operability determinations are sent to the Operations Department Manager for review and appropriate action. However, the acting operations manager requested a SORC review of the subject operability determination for the RCIC battery charger. SORC focused on the design bases and current licensing bases for the battery charger, asking specific questions regarding the function of the charger and its ability to perform the function while in a degraded condition. The questions led to NED revising the assessment to ensure that the appropriate design bases considerations were incorporated. On September 15, 1995, the RCIC battery charger was repaired by installing a temporary modification which interchanged the high voltage alarm card with the high voltage trip card. This change has eliminated the tripping phenomena because the alarm card has an internal filter which makes it less susceptible to the spiking condition.

The inspector assessed the licensee's actions regarding the degraded RCIC battery charger as good. The request by a SORC member to have SORC review the engineering assessment was excellent considering that the SORC review subsequently identified a number of concerns regarding design bases that had not been considered by the engineering organization. The SORC review resulted in a clear and understandable operability determination being provided to the plant operators. Subsequently, as a result of this improved information being available in the control room, the operators identified that the battery charger could not perform its required function, declared it inoperable and the temporary modification was installed to restore operability.

7.0 EXIT INTERVIEWS/MEETINGS

7.1 Resident Exit Meeting

The inspectors met with Mr. M. Reddemann 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.