U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No.	50-354/95-11
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:	July 9, 1995 - August 11, 1995
Inspectors:	R. J. Summers, Senior Resident Inspector S. A. Morris, Resident Inspector J. Furia, Radioactive Effluents Specialist J. Caruso, Reactor Operations Specialist

Approved: J. R. White, Chief, Reactor Projects Section 2A — 9/15/95 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.

EXECUTIVE SUMMARY

Hope Creek Inspection Report 50-354/95-11
July 9, 1995 - August 11, 1995

OPERATIONS

During this report period, operators committed a number of personnel errors, including mispositioning components which resulted in degraded performance of systems important to safety (e.g. residual heat removal, service water); failure to adequately control activities specifically prohibited by technical specifications (e.g. polar crane operation); and, failure to control safety tagging that could have led to personnel safety concerns. Operator performance during the startup from a forced outage, including procedural adherence, supervisory oversight, and communications was good. Subsequent power operations were conducted safely. An independent inspector review of the high pressure coolant system confirmed that it's configuration was adequately controlled and that it was capable of performing it's intended safety function.

MAINTENANCE/SURVEILLANCE

Maintenance and surveillance activities were generally effective at supporting reliable plant operation throughout the inspection period. A scheduled outage of the "C" emergency diesel generator was well controlled and coordinated and resulted in the equipment being returned to service ahead of schedule. Troubleshooting and root cause investigation into continued inoperability of the "A" control room emergency filtration system was well coordinated and resulted in an appropriate root cause analysis. Certain required surveillance activity deficiencies were identified by the licensee that led to a violation of plant technical specifications (see Section 3.2).

ENGINEERING

Good engineering support in the receipt and handling of new fuel was noted. A thorough engineering analysis of the potential damage to the "A" reactor recirculating water pump seal was developed to support plant operation following an error that resulted in the seal being pressurized by the control rod drive system. The design basis of the high pressure coolant injection (HPCI) alternate suction valve (1BJHV-F042) for the containment isolation function was left as an unresolved item to ensure that it did not need to be treated as a remote-manual isolation valve (see Section 4.2). A Non-Cited Violation resulted from engineering and QA review of a degraded HPCI minimum flow check valve (see Section 7.1).

PLANT SUPPORT

The radioactive waste system functional review was found to be extremely thorough and provided excellent, critical assessment of radwaste system operations. Response to the unusual high volume of radwaste was considered

(EXECUTIVE SUMMARY CONTINUED)

well coordinated with proper management attention. Condensate chemistry problems due to a breakdown of demineralizer resin contributed to the large volume of radwaste and impacted plant startup activities.

SAFETY ASSESSMENT/QUALITY VERIFICATION

Quality Assurance activities provided excellent, critical assessment of plant performance. The radiation protection program audit identified a number of strengths, as well as, a few weaknesses where improvement could be made; but, concluded overall that the program was effective. The routine Station QA report provided good assessment of the organizational response to the shutdown cooling bypass event. Other performance issues described in that report were consistent with independent NRC findings. Operations self assessment of the shutdown cooling bypass event indicated that procedure adherence, equipment failure, lack of a questioning attitude, and failure to adequately assess plant indicators all were root causes of the event. A Non-Cited Violation resulted from a recent QA audit of operations relative to diesel fuel oil storage tank level (see Section 7.1).

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DETAILS

1.0 SUMMARY OF OPERATIONS

The Hope Creek unit began the inspection period in a cold shutdown condition that was achieved on July 8, 1995, resulting from a forced manual shutdown in accordance with plant technical specifications due to a failure of the control room emergency filtration system. During this period, operators experienced a number of personnel errors, including mispositioning equipment which resulted in degraded performance of systems important-to-safety, and failure to adequately control activities specifically prohibited by plant technical specifications. While no cited violations resulted from these errors, one non-cited violation of the station's safety tagging program resulted. In addition, one of these errors involved mispositioned reactor recirculating water pump discharge isolation valves that resulted in seriously degrading the performance of the operating residual heat removal (RHR) system. This matter is the subject of an NRC special team inspection that was initiated on August 7, 1995 (see NRC IR 50-354/95-81).

Following the shutdown period, operators demonstrated good performance during the plant startup that occurred on July 25, 1995, and event free power operations were maintained throughout the balance of the inspection period.

2.0 OPERATIONS

2.1 Inspection Findings and Significant Plant Events

Except as noted in paragraph 2.2 below, the inspectors verified that Public Service Electric and Gas (PSE&G) generally operated the facilities safely and in conformance with regulatory requirements. The inspectors evaluated PSE&G's management control by direct observation of activities, tours of the facilities, interviews and discussions with personnel, independent verification of safety system status and technical specification compliance, and review of facility records. The inspectors performed normal and back-shift inspections, including 15 hours of deep back-shift inspections.

Operator Errors During the Shutdown Period:

During the inspection period a number of operator errors occurred while the plant was in a shutdown condition. These errors included: (1) on July 7, when securing a secondary condensate pump during the plant shutdown, an operator erroneously secured a primary condensate pump; (2) on July 8 to 9, when cycling reactor recirculating water pump discharge isolation valves for thermal binding considerations, an operator erroneously left two of the valves (BBHV-F031-A&B) partially open, resulting in degrading the operating residual heat removal (RHR) system by permitting required cooling water flow to bypass the reactor; (3) on July 11, operators erroneously commenced to release (lift) tags for work on the extraction steam system (tagout number 069100) while the work was still in progress, resulting in a violation of station tagging procedures and a potentially unsafe condition for maintenance personnel; (4) on July 12, when returning conditions to normal following a maintenance activity affecting the "A" recirculating water pump seal water line, an operator opened valve, 1BFHV-3800A, prior to opening either the recirculating water pump suction or discharge isolation valves, resulting in pressurizing the affected recirculating water pump seal package to control rod drive (CRD) system operating pressure; (5) on July 13, operators failed to provide positive control of the refuel floor polar crane as required by plant technical specifications when all four emergency diesel generators were declared inoperable; and, (6) on July 15, an operator erroneously tagged out-of-service the "B" service water pump traveling screen, resulting in the entire "B" loop of service water being technically inoperable, since the "D" service water pump was already inoperable. Each of these errors are discussed separately as follows:

Securing Condensate Pump During Shutdown

During the plant shutdown begun on July 7, an operator was in the process of realigning the condensate and feedwater systems per the integrated operating procedure. After reading the step to remove a secondary condensate pump from service, the operator erroneously secured an associated primary condensate pump. While this error was a result of failing to self-check per the licensee's STAR program, it did not result in any unusual transient condition for reactor vessel water level at that time.

Degraded Residual Heat Removal (RHR) System Operation

The Hope Creek unit was placed in the Cold Shutdown operating condition at 10:57 a.m. on July 8, with the "B" RHR system in service in the shutdown cooling mode of operation. Later, operators experienced difficulty in cycling open and closed the reactor recirculating water pump discharge isolation valves due to apparent thermal binding problems, and elected to leave these valves (BBHV-F031-A&B) partially open. This act was not approved by procedure and resulted in degrading the performance of the RHR system in both the functions to remove decay heat and provided sufficient reactor coolant system flow to ensure adequate temperature indication. The NRC held a management meeting with PSE&G on July 31, 1995 to discuss this event and the licensee's preliminary findings of their own investigation. The presentation materials used at that meeting are attached to this report.

As a result of this event, the NRC dispatched a special inspection team to the site to independently verify the causes of the event and assess the consequences. The NRC will review the results of the special team inspection to determine if any enforcement action is warranted in this matter. (See NRC IR 50-354/95-81 for additional detail.)

Extraction Steam Tagging Error

PSE&G identified a tagging incident on July 11, 1995, involving tagging request number 069100. This tagging request was issued to make repairs to feed water heater 6A drain valve, 1AF-V045. The control room supervisor signed the release authorization for the tagging request. Consequently, protective tags were removed with work still in-progress on the system.

The tagging request initially only identified one work activity (i.e., work request number 950505189) as being outstanding but in fact there were two other outstanding work activities (i.e., work requests 980925001 and

940510151) that were not identified on the tagging request at the time. These two other work activities were utilizing this tagging request to provide tagging and isolation protection for work in-progress. The control room supervisor was not aware of these other outstanding work activities when he signed the release authorization for the tagging request. The problem was discovered when a maintenance supervisor subsequently attempted to sign onto the tagging request for his work activity which was one of the two work activities not listed on the tagging request. Apparently, the operations department work control group had not maintained the tagging request updated by listing all outstanding work activities on the request as required.

This equipment control problem, which had the potential to result in personnel injury or equipment damage, was discussed by the Operations Manager with the entire management staff. Subsequently, work control personnel were provided clear guidance to ensure all work requests would be listed on the tagging request. In addition, individual supervisor names were assigned to the tagging request in lieu of titles, in order for the control room supervisor to ensure all activities were completed prior to releasing the tags.

Reactor Recirculating Water Pump Seal Water Error

On July 12, 1995, an operator erroneously placed the control rod drive (CRD) hydraulic system seal water supply to the "A" reactor recirculating water pump in service with the seal return isolated. An operator aid on the control panel warned operators against such action because seal pressure would rapidly rise to CRD system pressure. No damage to the equipment apparently resulted. This matter is discussed further in paragraph 3 of this report.

Inadequate Control of the Polar Crane

On July 13, 1995, at approximately 2100, the operations shift declared all four emergency diesel generators inoperable because of a self-identified failure to adequately implement the technical specification surveillance requirements that demonstrate the operability of vital bus load shedding circuitry in response to a loss of offsite power. The operators appropriately entered the technical specification 3.8.1.2.a action statement for A.C. sources required while the unit was in operating mode 4. This specification requires, in part, that with less than two diesel generators operable, crane operations over the spent fuel storage pool shall be suspended when fuel assemblies are stored within.

The inspectors attempted to verify that all of the associated action requirements for the above noted specification were promptly and properly implemented. With respect to the prohibition on crane operations, the inspectors determined that the polar crane was administratively controlled by means of a "worker blocking tag." Use of such a tag permits an individual, specifically stated on the tag, to operate equipment within the tagging boundary in order to complete maintenance activities. A worker blocking tag does not positively control the status of a tagged component under the direction of operations department personnel. However, the crane was tagged to support the completion of annual preventive maintenance.

The inspectors found no information to conclude that the crane had been moved during the period of time which the technical specification action requirement applied (approximately four days). However, the work order that authorized the crane maintenance did not contain any written restrictions regarding crane operation. Further, the individual listed on the blocking tag was not aware of the technical specification requirement. Because of the apparent lack of positive control over the polar crane by operations, the inspectors concluded that it was only fortuitous that no crane operation over the spent fuel storage pool occurred.

Service Water Tagging Error

On July 15, 1995, during a walkdown of a tagging operation on the "D" service water system pump traveling screen prior to commencing scheduled maintenance, a tagging error was identified by the licensee. An operator erroneously tagged out-of-service the "B" service water system traveling screen instead of the "D" screen. This resulted in both trains of the "B" service water loop being technically inoperable. The error was immediately corrected by plant operators, restoring configuration to normal. While the "B" traveling screen was out-of-service, the "B" service water system pump remained in service and performance was unaffected. However, since the "B" loop of service water was providing the heat sink to the operating RHR shutdown cooling system at the time, there was potential for this event causing a loss of shutdown cooling.

The licensee immediately corrected the configuration control problem; counseled the operator who tagged the wrong component; and, subsequently determined that the Tagging Request Information System also failed to identify the traveling screen circuit breaker as needing second verification, which was also corrected.

NRC Assessment of Operator Errors

All of the above described errors involve operational configuration control. In addition, most of the errors involve procedure adherence problems, or a failure to implement other explicit guidance. The errors resulting in the degraded shutdown cooling system will be discussed further in NRC inspection report 50-354/95-81. The failure to ensure that the polar crane was not used as required by the plant technical specifications did not result in a violation only because the crane was not in use at the time. The other errors all had potential to harm station personnel or result in degraded safety systems. The licensee has included the above events in a special review of operator errors. This was a committed action at the management meeting held on July 31, 1995. These issues will be assessed in a future NRC inspection as part of the followup on corrective actions taken for the shutdown cooling bypass event (URI 50-354/95-11-01).

2.2 Monitoring of Ultimate Heat Sink Level and Temperature

During a routine control room observation, the inspectors noted that neither the ultimate heat sink (Delaware River) level or temperature indications were functioning properly; that is, they were both marked as inoperable. The inspectors questioned control room operators and supervisors about how these important technical specification required parameters were being monitored. Both of these parameters were of particular interest this report period since river temperature approached 85 degrees F (requiring increased monitoring frequency per license requirements) and river level exceeded 95 feet above mean sea level requiring entry into an operations department abnormal procedure for site flood protection. Hope Creek operations personnel stated that these indicators have had a long history of reliability problems and that the required measurements were obtained from Salem station operators via telephone. Operators were also able to use station service water inlet temperature indication from the control room information display system (CRIDS) as an alternate.

The inspectors learned that both of these faulty indicators were being tracked as either operator work-arounds or as engineering department action items. In fact, late in the period, a modification was installed for the river level indicators that was intended to correct long standing problems associated with these devices. Upgrades were also being planned for temperature monitoring. However, despite recent efforts to improve system performance, the inspectors concluded that the long term reliance on alternate sources of river temperature and level indication was indicative of a lack of aggressive followup and resolution to known problems.

2.3 Observation of Unit Startup

The inspectors observed portions of the unit startup that occurred July 25, 1995. Specifically, the inspectors witnessed the latter phases of reactor coolant system heatup, the subsequent power increase and the turbine generator warmup and synchronization to the offsite electrical distribution network. Though the startup was complicated by several minor equipment deficiencies, including spurious trips of the "B" reactor feed pump turbine (off line during the startup), a malfunctioning turbine auxiliaries cooling system temperature control valve, and a failure of the "2-6" breaker in the 500 KV switchyard, the inspectors concluded that an appropriate focus on safe operation of the unit was maintained. Good procedural adherence, attention to detail and supervisory oversight were also observed. Effective use of three way communications was observed. However, on one occasion, the shift supervisor was noted to manipulate controls on turbine generator support equipment during troubleshooting, which potentially could have distracted him from his primary role.

2.4 High Pressure Coolant Injection System Walkdown

During the report period, the inspectors conducted a walkdown of the accessible portions of the high pressure coolant injection system, including both mechanical, electrical, and control function components. At the time of the review, the unit was in operating condition 4, (cold shutdown), resulting in the system being out of its normal standby lineup for emergency core cooling. Further, corrective maintenance was being performed on various support systems at the time of the walkdown.

The inspectors independently confirmed that system valve and electrical lineups matched the tagging request information system current status. Further, as-built system configuration was compared to plant drawings and found to be consistent. Technical specification requirements and final safety analysis report descriptions were reviewed against PSE&G operating and surveillance procedures and no concerns were identified. Housekeeping in the vicinity of system components was good. Finally, no degraded conditions associated with system controls or indications were noted. Within the scope of this review, the inspectors concluded that the high pressure coolant injection system was capable of performing its intended safety function.

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 PSE&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. The inspectors more closely evaluated the activities in section 3.2 below.

3.2 Inspection Findings

Emergency Bus Undervoltage Logic Circuitry Testing

After reviewing the existing surveillance testing procedures used to perform the load shed testing for the safeguard loads on the emergency buses, PSE&G issued condition report (CR) number 950713217 on July 14, 1995 that concluded that technical specification (TS) testing requirements 4.8.1.1.2.h.4a and 6a had not been verified by existing testing and as a result declared the emergency diesel generators inoperable. The testing review was in response to logic configuration discrepancies identified by CR number 950703184 issued on July 3, 1995.

The omitted testing did not provide complete verification of the circuits associated with load shedding on the vital buses in response to a loss of normal offsite power (LOP) signal. Testing deficiencies were found in contacts 3-4 (i.e., continuity checks of the 27X degraded feeder contacts) and 7-8 (i.e., bus voltage available contacts for start permissive for the RHR and core spray 4160 volt circuit breakers) of relays 27AY1 and 27AY2 and all contacts of the 27AX1 and 27AX2 relays (i.e., that enable load shed of 4160 volt breakers for SACS, core spray, RHR, SSWP, and the chiller initiated by the bus loss of voltage relays). PSE&G subsequently performed specialized testing that satisfied technical specification requirements with no identified deficiencies.

The inspectors concluded that the plant procedures did not adequately test the bus undervoltage logic and the failure to perform the load shedding testing is an example of a violation of technical specification 4.0.2 for failing to implement the requirements of technical specifications 4.8.1.1.2.h.4.a and 6.a.

NRC Information Notices 91-13, "Inadequate Testing of Emergency Diesel Generators" and 92-40 "Inadequate Testing of Emergency Bus Undervoltage Logic Circuitry" were issued to alert licensees of a safety problem that could result from the use of an undervoltage test method that fails to verify the de-energization of the emergency safety buses. The NRC will review the licensee's operations experience feedback program handling of these NRC generic communications as part of the followup to this issue to ascertain why corrective actions were not taken as a result of that information.

Inadequate Functional Testing of ECCS Actuation Instrumentation

On July 20, 1995, during a work order completion review of an emergency core cooling system actuation channel calibration, PSE&G discovered that the associated maintenance procedure (HC.IC-CC.BC-033 (Q)) incorrectly allowed credit to be taken for a channel functional test as required by technical specification 4.3.3.1, ECCS Actuation Instrumentation. The individual performing the review identified that the channel calibration procedure tested only a single analog trip unit in the Automatic Depressurization System (ADS) permissive circuitry, and that the functional test was intended to test three analog trip units. These tests were required technical specification surveillance requirements to demonstrate the operability of the associated actuation logic instrumentation.

PSE&G subsequently determined that the above example was not an isolated case. In fact, prompt followup review of the concern led to the discovery that nearly all other ECCS actuation channel calibration procedures allowed credit for the associated functional testing, but that this credit was only a problem when there were multiple analog trip units associated with a functional test. Further review led to identification that at least three functional tests had been inappropriately credited in the past and were overdue by technical specification definition. One example, regarding testing of the high pressure coolant injection suction transfer logic, should have resulted in the injection system being declared inoperable. The other two examples involved inadequate testing of permissive logic for the ADS from both core spray and residual heat removal subsystems.

PSE&G initiated short term corrective actions which included the satisfactory performance of all the identified overdue surveillances. Further, procedure revisions were generated to correct the noted errors. Long term actions planned included a detailed root cause analysis, and a thorough search of work order history to determine how long this issue had existed at the station. The inspectors concluded that this problem, though self-identified and promptly resolved, represented weak performance in the area of surveillance implementation, and as such constituted a second example of a violation of technical specification 4.0.2 requirements for failing to implement the requirements of technical specification 4.3.3.1 **(VIO 50-354/95-11-02)**.

Emergency Diesel Generator Outage

On August 7, 1995, Hope Creek commenced a scheduled outage of the "C" emergency diesel generator in order to perform technical specification surveillance activities, specifically the 18 month engine maintenance requirement in accordance with vendor recommendations. Hope Creek was recently issued license changes that increased the allowed outage times for the "C" and "D" diesels from 72 hours to 14 days and reduced the amount of engine maintenance necessary to satisfy the surveillance requirements. These amendments enabled Hope Creek to perform this work on-line as opposed to during unit outages.

The inspectors observed portions of the work during the diesel outage as well as the administrative controls in place to ensure the maintenance was carried out safely and effectively. Technical specification 3.8.1.1 action requirements were appropriately implemented, including the timely execution of the offsite power distribution lineup verifications. The net safety gain analysis performed to justify conduct of the work while the unit was on line was thorough and considered probabilistic risk data in the assessment. The inspectors noted that there were comprehensive pre-job training sessions with maintenance technicians to ensure the work would proceed efficiently. During the outage, the inspectors noted excellent coordination of work activities, including oversight by the system outage manager, maintenance supervision, and system engineering. Vendor personnel also contributed effectively.

At the conclusion of the outage, all diesel post maintenance tests were completed satisfactorily with only minor complications. Further, the diesel generator was restored to an operable status ahead of schedule. The inspectors concluded that Hope Creek did an excellent job of controlling, supervising, and restoring from the outage of this important safety system.

Troubleshooting Control Room Emergency Filtration System

The inspectors assessed the troubleshooting of the "A" control room ventilation chiller unit of the control room emergency filtration system following the technical specification required plant shutdown on July 7, 1995. The inspector concluded that PSE&G's recent troubleshooting activities were thorough, well coordinated and well managed; and resulted in reasonable root cause analysis. Nonetheless, these chillers have been the subject of numerous PSE&G incident reports, licensee event reports and special reports dating since 1986, which is indicative of weakness in failure analysis and ineffectiveness of corrective actions.

At the time of the plant shutdown there were numerous failure hypotheses for the data but the exact nature of the failures or the location of the fault could not be determined. PSE&G believed that, due to the design of the system, there were several process trips which did not provide a unique indication as to the trip. Additionally, there were many components which could have failed on an intermittent basis without providing a clear, identifiable record of the source of the failure. Prior to shutting down the plant, PSE&G requested that the NRC grant enforcement discretion and relief from their technical specification relative to allowed outage time in order to continue troubleshooting efforts. This request was denied since PSE&G was unable to determine the root cause of the problem after performing extensive troubleshooting.

The inspectors reviewed some of the performance history for the CREF system to determine if there were unique factors affecting the system and to ascertain the effectiveness of prior licensee troubleshooting and maintenance activities. The inspectors found that the safety related "A" control room ventilation chiller unit (1AK400) had a history of tripping much more than the "B" chiller unit (1BK400). The inspectors also reviewed a report (HSR-93-004, dated January 29, 1993) issued by the onsite safety review engineer that documented twelve control room ventilation failures that resulted from problems related to the chillers between 1986 and 1992. Tripping frequency for the "A" chiller increased recently and resulted in the seven day time limit for technical specification 3.7.2 on control room habitability being reached, forcing an outage of Hope Creek on July 8, 1995. Eight trips of the "A" chiller had been experienced between June 24 and the plant shutdown on July 8, 1995.

PSE&G formed a multi-disciplined group to troubleshoot the problem and to determine the root cause. The team was led by the system engineer with close oversight from the site engineering manager, and included instrumentation and control (I&C) engineers and technicians and a private consultant specializing in failure prevention and investigations. Outside assistance from the equipment vendor was also utilized.

Troubleshooting efforts involved a number of different activities. Data recording equipment was attached to the chiller circuitry after the trip on June 30, 1995 and captured data from all subsequent trips (i.e., except the July 7 trip). Additional monitoring instrumentation and data recorders were later installed to further identify the cause of the trip. In addition, extensive inspections and system walkdowns were conducted to determine if any components showed signs of abnormal operation or wear. All major components were inspected and no obvious component failures were detected that could be associated with the trips. The chiller circuitry was also checked for loose connections and possible vibration-related intermittent grounding using time domain reflectometry.

PSE&G determined that additional cycling loads on the bus powering the "A" chiller as well as longer cable lengths on the "A" chiller control circuitry accounted for significantly larger voltage drops across the "A" chiller when compared to the "B" chiller. Voltages in various parts of the circuitry were measured and it was found that voltage could be significantly affected by cycling loads. In the control room, 25 amp swings were noted on the bus feeding the "A" chiller and about half that much on the bus feeding the "B" chiller. In addition, cable lengths to the Bailey controllers on the "A" and "B" chillers were measured and "A" chiller had cable lengths about double the lengths of the "B" chiller (i.e., "A" cable lengths were approximately 2,000 feet longer). This difference in cable length between the two chillers was determined to cause a voltage drop of 11-13 volts across the control circuitry on the "A" chiller as compared to about a 5 volt drop on the "B" chiller.

Seventeen scenarios were postulated to explain the chiller trips. Each scenario was examined using available data for supporting and refuting evidence. Fifteen of the seventeen scenarios were refuted and the chiller was tested by purposely introducing a momentary power interruption (less than one cycle). The hope was that the system would trip again to definitely isolate the cause to a specific component. The chiller tripped after several repetitions and the recorder traces taken were found to be equivalent to those traces recorded from earlier trips of the chiller.

The PSE&G troubleshooting team concluded that the root causes of the "A" chiller trips included lengthy cable runs (not considered during initial design) and power source voltage drops outside the chiller (due to transient loading of the bus combined with the inrush current needed to re-energize numerous relays) which had resulted in degraded voltage in the control circuit. The system had been unable to recover from such degraded voltage conditions causing the chiller to trip. The design of the circuit was intended to allow up to 0.2 seconds of power interruption, but was inadequate for this purpose. Possible thermal aging effect or power source effects on the Robert Shaw modules were investigated as a contributing root cause.

A design change package was developed and installed which effectively reduced the cable lengths by installing interposing relays. The measured voltage drop across the "A" chiller control circuit went from 11-13 volts before the modification to about 5 volts after the installation of the modification. Post modification testing verified the circuit's ability to tolerate momentary interruptions or dips in control power voltage.

Based on continued control room ventilation failures that resulted from problems related to the chillers between 1986 and the most recent failures, the inspectors concluded that previously implemented corrective actions were not sufficient to preclude recurrence of further problems. In general, the root cause of previous failures was not comprehensive enough to identify all potential failure contributors. PSE&G concluded that based on the extensive analyses performed, the recently installed design changes should dramatically improve reliability. In addition, a project team was established to address long term corrective actions that could further enhance the reliability of the chillers. The inspector indicated to licensee management that the effectiveness of the corrective actions would be determined based on the future demonstrated performance of the modified system.

3.3 Followup of Prior Inspection Findings

(Closed) Unresolved Item 50-354/93-25-02 Inadvertent RHR Pump Start

In NRC inspection report 50-354/93-25, the inspectors documented an event which occurred on November 9, 1993 involving an inadvertent start of the "B" residual heat removal pump. The pump start was caused by a maintenance technician performing a surveillance test of emergency core cooling system initiation logic. Subsequent licensee investigation revealed that the technician used an internally shorted logic tester which, when placed in the circuit under test, caused the pump start signal to be generated. As a result of this event, the inspectors questioned whether these logic testers could be checked for internal faults prior to use in safety related circuits.

During this report period, the inspectors reviewed PSE&G's corrective actions resulting from the above noted event. Actions included revisions to maintenance procedures that necessitated use of the subject test equipment, specifically requiring technicians to verify proper mode switch positions by use of a digital multimeter prior to installing the tester in logic circuits. Further, routine logic tester overhauls were performed to ensure active components internal to the tester were replaced on a periodic basis before they could degrade. Finally, and most significantly, Hope Creek technical department personnel designed a new logic test device that relied only on passive internal components and simplified logic test procedures. No further problems have been experienced with these test devices since these corrective actions have been implemented.

The inspectors concluded that Hope Creek effectively diagnosed and remedied the problem with the emergency core cooling system initiation logic testers. Based on this review and assessment, this item is closed.

4.0 ENGINEERING

4.1 Inspection Findings

New Fuel Receipts

During the report period, Hope Creek began receiving new fuel shipments in preparation for the upcoming refueling outage scheduled for November 1995. The inspectors observed all facets of the fuel receipt process, including truck delivery, uncrating, inspection, channeling, and transfer to the storage pool and vault. Radiation protection practices were also reviewed and assessed. The inspectors noted that appropriate procedures for control of the evolution were available and current, and were adhered to explicitly. Good coordination between reactor engineering, maintenance, and radiation protection personnel was evident and resulted in effective completion of the work without any identified problems. No problems regarding compliance with new fuel shipment and handling license requirements were observed.

Two problems occurred that resulted in technical support followup. The first involved the inadvertent dropping of a security badge and dosimeter into the spent fuel storage pool by an individual performing work on the refueling bridge. The badge was not able to be retrieved. The second concern involved apparent oil leakage from an air motor on one of the fuel preparation machines that resulted in contamination of the fuel pool. This leakage was quickly identified and terminated. Good followup by technical department personnel ensured that corrective actions were appropriate. The pool contamination event resulted in the newly received fuel being transferred to the new fuel vault, which ultimately will force the fuel to be handled again in order to place it into the pool for refueling. The inspectors concluded that, in spite of these two issues, the new fuel receipt process was well controlled and implemented.

Potential Overpressurization Reactor Recirculation System

An action request (AR) number 950712209 was written to identify a potential overpressurization of the reactor recirculation piping, pump, and seal package due to an operator error. The operator had valved in CRD seal pressure to the "A" recirculation pump with the pump isolated.

The inspectors concluded that the "A" recirculation pump seal high pressure condition was appropriately evaluated. The inspectors reviewed problem report number 950713095, which evaluated the effects of subjecting the "A" recirculation pump seal to a high pressure condition. The inspectors reviewed the system engineer's evaluation that concluded that the seal was subjected to 1250 psig and should not have been damaged based on pump design requirements. The seal vendor was consulted and concurred with the system engineer's conclusion that the seal should not have been damaged. Operators ran both recirculation pumps prior to reactor start up to ensure the seals were performing as designed. Additionally, operations and system engineering have been closely monitoring seal performance since plant startup, and have noted no abnormalities associated with the seal performance.

Recirculation System Runback During Shutdown July 7, 1995

While removing the "B" reactor feed pump (RFP) from service during shutdown (with "C" RFP out of service and "A" RFP in service), total feed flow momentarily dropped below 20% due to sluggishness of the "B" RFP. This caused a full runback to occur on the "B" recirculation pump. The full runback did not occur on the "A" recirculation pump. Full runback should have occurred on both "A" and "B" recirculation pumps.

The inspector concluded that the troubleshooting activities for resolution of the "A" recirculation runback were appropriate and timely. Corrective maintenance work order number 950709072 was initiated on July 9, 1995, to troubleshoot and repair the "A" and "B" recirculation runback control loop. Time delay relay K23A (B31) was found to be out of calibration. The relay was set at 22.5 seconds vice about 15 seconds, which caused the "A" recirculation runback not to occur. The relay was replaced, calibrated and retested.

However, the inspectors questioned the delayed response to the feed pump performance, which had been the initiating event. The system engineer initiated AR number 950714122 on July 14, 1995, one week after the event occurred and after the inspector raised a concern that the problem had not been previously documented in any corrective action system record. The system engineer indicated that troubleshooting could occur in parallel with the scheduled plant startup if necessary since the feed system is a balance-of-plant and a non-technical specification system. The inspector noted that the resolution of this problem and root cause determination was not highlighted as a priority activity until after the inspectors questioned the engineers involved with this matter.

4.2 Followup of Prior Inspection Findings

(Closed) Unresolved Item 50-354/94-11-01 High Pressure Coolant Injection System Alternate Suction Valve 1BJHV-F042

On June 9, 1994, PSE&G discovered that the high pressure coolant injection system alternate suction valve (1BJHV-F042) did not meet the commitments stated in the current licensing basis. This valve provides an injection system suction flow path from the suppression pool. The unresolved issue, which focused on how PSE&G addressed the NUREG 0737 requirement for automatic isolation valves, was thoroughly described in NRC inspection report 50-354/94-11. The concern was that the valve would automatically reopen following reset of a system isolation signal, contrary to the commitment made in the final safety analysis report.

During this report period, the inspectors reviewed a letter from PSE&G, dated August 23, 1994, which clarified the concerns regarding the automatic controls for the suppression pool suction valve. Specifically, the letter justified that this valve need not meet TMI Action Item II.E.4.2 requirements relative to required operator action to reopen the valve following automatic isolation. Further, the final safety analysis report was updated with the new information. The inspectors concluded that, based on this documented information and a review of system design requirements, PSE&G adequately addressed the issue of the licensing commitment made in response to the NUREG 0737 requirement. This item is closed.

Notwithstanding, the inspectors questioned why the 1BJHV-F042 suppression pool suction valve was listed as an automatic primary containment isolation valve in technical specification Table 3.6.3-1.A.5(b). The inspector noted that it was not desirable for a primary containment isolation signal to cause this valve to isolate, since the high pressure coolant injection system is credited in the Hope Creek accident analysis for loss of coolant accident scenarios. Further, all other emergency core cooling systems, as well as the reactor core isolation cooling system, have suppression pool suction valves that are listed and controlled as remote manual containment isolation valves. In addition, the inspectors questioned whether the 1BJHV-F042 valve had appropriate control logic or was tested in accordance with appropriate surveillance criteria. These issues will remained unresolved pending further review by PSE&G and NRC personnel **(URI 50-354/95-11-03)**.

5.0 PLANT SUPPORT

5.1 Radiological Controls and Chemistry

The inspector periodically verified PSE&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 Waste System Functional Review

During this report period, PSE&G completed an extensive functional review of the radioactive waste processing system. A multi-disciplined team of twenty auditors was assembled to conduct the review, which was modeled after the NRC's safety system functional review inspection program module. On July 28, 1995, the Hope Creek safety review group led team held an exit meeting to announce its major findings and present its assessment.

The inspectors attended the exit meeting and reviewed the findings identified by the team. The inspector noted that many of the findings were significant and that recommended corrective actions appeared appropriate. Further, all of the concerns identified that required followup activity were entered in the station's action tracking system in order to ensure that they would be prioritized and tracked to completion. The inspectors concluded that this system functional review was extremely thorough and well managed, and should ultimately result in improved radioactive waste processing system performance and reliability.

Radioactive Waste System Concern

During this inspection period, a specialist inspector visit was conducted to determine if there were any outstanding, unaddressed deficiencies affecting the radwaste handling and storage systems. System contaminations resulting from the April release event, and systems taken out-of-service due to chemistry problems resulting from the degradation of the condensate demineralizers, required that HVAC condensate be processed as radwaste. Additionally, difficulty was experienced in the ability to process high conductivity water through any of the waste evaporators or the crystallizer, transfer spent condensate demineralizer resins for use in the radwaste deep bed demineralizers, or effectively process the increasing volume of radioactive water in the reactor building sumps. In response to these conditions, the licensee initiated projects to address each of these issues. These activities included contracting with Chem Nuclear Systems, Inc. to process high conductivity water; isolating system leaks to reduce the activity found in the reactor building sumps; initiating actions to permit clean HVAC condenser water to be processed as clean waste; and, increasing management attention toward efforts to establish effective operation of radioactive waste processing facilities.

5.2 Emergency Preparedness

The inspector reviewed PSE&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 PSE&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 PSE&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 PSE&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

Quality Assurance Audit Report 95-150 was issued on July 14, 1995. This report reflected the results of the 1995 Nuclear Business Unit Radiological Protection (RP) Program. The audit was conducted in June 1995 and concluded that the radiation protection program was being effectively implemented. A number of strengths were identified, including: RP Self-Assessment training; timely and effective corrective actions; focus on improvement; good communications; clear goals; supervisor involvement in training; effective engineering controls; high quality ALARA instructions; and, good radiation worker job briefings. Four areas for improvement were identified for Hope Creek, including: several procedure adherence problems relative to Radiological Occurrence Report procedure implementation; supervisor qualification records not properly administered; an inappropriate action by an RP technician in response to a personnel exit ALNOR alarm condition; and, a failure to conduct a 10 CFR 50.59 review regarding RP organizational changes.

Quality Assurance Monthly Report - July 1995 was issued on August 2, 1995. The report described QA observations and assessments in the areas of operations, maintenance, engineering and plant support. In the operations area, the report had a number of findings, the most significant of which included actions relative to the RHR shutdown cooling bypass event of July 8 and 9, 1995. In summary, the report concludes that both QA and the Safety Review Group (SRG) ". . . assessed the event evaluation and continued to express a concern that the organization's response was inappropriate for the significance of the event." The report describes the sequence of communications with the Hope Creek organization that ultimately led to a recommendation to station management to form a multi-disciplined team to review the event. This recommendation was implemented by the General Manager - Hope Creek Operations on July 20, 1995. Other findings included: generally good implementation of various maintenance activities observed; some minor housekeeping problems that were immediately corrected; effective implementation of the motor operated valve inspections for overgreasing; and, followup for corrective actions for a concern relative to material (piping) identification and traceability identified by the maintenance department.

On August 2, 1995, the General Manager - Hope Creek Operations announced the development of the Hope Creek Performance Improvement Action Plans. The action plans will include overall station goals and specific strategies and actions to achieve the goals within the general framework provided by the Nuclear Business Unit Impact Plan. Immediate and near-term actions are expected to be completed by the end of 1995. Long-term actions will be completed by the end of 1996. A team, consisting of various plant staff and management, including contractor assistance from PRISM, was formed to identify performance issues and barriers to future improvements.

On August 10, 1995, the acting Operations Manager announced the findings of the operations department review of the personnel error that led to the RHR shutdown cooling bypass event. That assessment concluded that the root causes of the event included: thermal binding of valve HV-F031-A and failure of the motor operator of valve HV-F031-B; procedure non-compliance by the plant operators in throttling open the HV-F031 valves; lack of questioning attitude on the part of the plant operators; and, failure to believe and accurately assess available indication by the plant operators. In addition, lack of specific training and failure to request technical support contributed to the causes. The assessment further described that adequate core cooling was assured at all times and that no adverse radiological consequences resulted. While the event consequences were minimal, the event was considered significant due to the personnel errors and the failure to recognize the plant heatup. Further, it was identified that station management faile) properly evaluate the event significance and communicate such to the NRC.

The inspectors periodically attended the Station Operations Review Committee (SORC) meetings to ensure the technical specification safety review requirements were being implemented. On all occasions observed, appropriate attendance (quorum) and chairmanship were maintained. The discussion focus was on the safety impact to the plant. Recently, the General Manager - Hope Creek Operations implemented an organization change that affected the voting membership of SORC. The Chemistry Manager was reassigned full-time duty leading a team review of technical specification surveillance requirement implementation. The acting Chemistry Manager, K. Maza, will become the voting member on SORC, until such time that a permanent Chemistry Manager is named. In addition, a new assistant Radiation Protection Manager was named, T. Cellmer, who will serve as the voting member on SORC. Additional changes were made regarding non-voting member attendance at SORC meetings to ensure that station QA and Licensing and Regulation organizations are represented. The NRC inspectors noted that SORC technical specification requirements were met during this inspection period. Further, the inspectors assessed that generally good safety assessments were performed by SORC, with appropriate critical evaluations of station activities.

Except for the issues associated with the shutdown cooling bypass event, the NRC inspectors assessed that the licensee's safety assessment and quality verification activities were timely, accurate and sufficient to provide necessary safety review of important station activities. The NRC is still evaluating the organization response, including adequacy of safety assessment and quality verification, relative to the shutdown cooling bypass event.

7.0 LICENSEE EVENT REPORTS (LER), PERIODIC AND SPECIAL REPORTS, AND OPEN ITEM FOLLOWUP

7.1 LERs and Reports

The inspectors reviewed the following LERs to determine whether the licensee accurately described the event and to determine if licensee responses to the events were adequate.

Number	Event Date	Description
LER 95-007	June 3, 1994	Diesel fuel oil storage tank found below technical specification minimum level requirement.

This LER describes a design deficiency with the diesel fuel oil storage tank (DFOST) level alarm and indication systems. The deficiency resulted in an inoperable emergency diesel generator in June 1994. At the time, an operability determination was made that incorrectly determined that the diesel was operable. Upon further evaluation during the recent QA audit of Hope Creek operations, this event was reassessed and determined to have resulted in a technical specification violation. Also, since the operability determination was faulty at the time, the event was not reported to the NRC as required. The inspector reviewed the licensee's corrective actions and found that they were comprehensive. While the design deficiency has yet to be corrected, interim measures have been established to provide better control of the DFOST level. This licensee identified and corrected violation of technical specifications and NRC reporting requirements is being treated as a Non-Cited Violation, consistent with Section VII of the NRC Enforcement Policy.

LER 95-008	June 8, 1995	Unplanned inoperability of a single train safety system - high pressure coolant injection (HPCI)

This LER describes an event resulting in the HPCI system being inoperable. An on-the-spot change was found to lead to a condition rendering the HPCI system inoperable due to isolating the HPCI turbine oil cooling water flow path during certain specified test conditions. This event was described in NRC inspection report 50-354/95-10, section 2.1, as a non-cited violation. The NRC inspector reviewed the corrective actions in the LER and found that they were accurate and comprehensive.

LER 95-009 June 13, 1995 Technical specification violation - surveillance requirements for the explosive squib valves for the Traversing In-core Probes

This LER describes an event resulting in missed technical specification required surveillance activities for the explcsive squib valves (containment isolation valves) for the Traversing In-core Probe system. This matter was reviewed in NRC inspection report 50-354/95-10, section 3.2, and a violation was assessed.

LER 95-010 June 21, 1995 Technical specification violation - improper entry into high radiation area

This LER describes an event resulting in a violation of plant radiation protection procedures and technical specifications when workers made an improper entry into a high radiation area in order to perform equipment maintenance. This matter was documented in NRC inspection report 50-354/95-10, section 5.1, and a non-cited violation was assessed.

LER 95-011 June 21, 1995 Technical specification violation - failed to implement action requirements of technical specification 3.4.8 and an unplanned inoperability of a single train safety system - high pressure coolant injection (HPCI)

This LER describes a series of events leading to an inaccurate operability determination based on "engineering judgement" relative to a small leak on the HPCI minimum flow check valve. After further review by QA and system engineering, the valve was subsequently isolated, as required, rendering the HPCI system inoperable. This event is also discussed in NRC inspection report 50-354/95-10, section 2.1. The inspectors reviewed the licensee's corrective actions and found that they were appropriate. This licensee identified and corrected violation of technical specifications is being treated as a Non-Cited Violation, consistent with Section VII of the NRC Enforcement Policy.

LER 95-012 June 26, 1995 Technical specification violation - unlocked high radiation door

This LER describes an event resulting in an improperly secured door in violation of the plant radiation protection procedures and the technical specifications. This event was described in NRC inspection report 50-354/95-10, section 5.1, and a non-cited violation was assessed.

The LERs listed above are considered closed.

7.2 Open Items

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

Number	Report Section	Status
URI 354/94-011-01	4.2	Closed
URI 354/93-025-02	3.3	Closed

8.0 EXIT INTERVIEWS/MEETINGS

8.1 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.

8.2 Specialist Entrance and Exit Meetings

Date(s)	Subject	Inspection Report No.	Reporting Inspector
August 7-11, 1995	Special Team Inspection	50-354/95-81	J. Trapp

8.3 Management Meetings

- July 31, 1995 Management meeting in NRC Region I Office - Shutdown Cooling Bypass Event
- August 3, 1995 Hope Creek SALP Management Meeting at PSE&G Processing Center
- August 24, 1995 Shutdown Cooling Bypass Event NRC Special Team Inspection Public Exit Meeting at PSE&G Processing Center

8.4 Licensee Management Changes

Effective July 31, 1995, Mr. Joseph Pollock became the QA Manager for Hope Creek. Mr. J. DeFebo, who held the position of Acting QA Manager returned to the position of Plant Support Assessment Supervisor for Hope Creek.

ATTACHMENT

JULY 31, 1995 MEETING PRESENTATION MATERIALS

MANAGEMENT MEETING ON JULY 8, 1995 SHUTDOWN COOLING BYPASS

AGENDA

INTRODUCTION	M. Reddemann
DISCUSSION OF EVENT	C. Bauer
CORRECTIVE ACTIONS	H. Hanson
CONCLUSION	M. Reddemann

MANAGEMENT MEETING ON JULY 8, 1995 SHUTDOWN COOLING BYPASS

INTRODUCTION

- o Overall Response (Event specifics to be presented later)
  - Event occurred over two shifts on July 8 and 9 during unit shutdown. Upon recognition, action requests were initiated for the recirc valve failure and the operator response to the event.
  - Due to the seriousness of the event, several other immediate actions were taken.
  - Upon return, GM formed an independent review team to conduct a thorough review of the circumstances surrounding the event.
  - Charter was developed. Team composed of individuals from:
    - Engineering
    - Training
    - Maintenance
    - Safety Review Group
    - Emergency Preparedness
    - Peer from Peach Bottom
  - Team started on July 20
  - Initiated a second root cause team to take a broad look at the recent increase in operator errors and those that occurred since the last refueling outage.
  - We will modify our operator training programs to capture the lessons learned from the event.
  - It is the GM's expectation that independent review teams be initiated immediately after a significant plant event

SHUTDOWN COOLING BYPASS EVENT REVIEW TEAM CHARTER

Conduct a thorough and systematic review of the circumstances surrounding the shutdown cooling bypass event at Hope Creek on July 8, 1995, and the resulting increase in reactor coolant temperature. The general objectives of the team are to:

1. Evaluate the event for a potential mode change.

2. Evaluate the event for a potential loss of shutdown cooling function. Determine if shutdown cooling path was esta lished in accordance with Technical Specifications and if temperature indication was adequate.

3. Evaluate the event for potential NRC reportability.

4. Assess the organization's, i.e., management's, response to the event.

5. Examine any equipment failures that contributed to the event and identify root causes.

6. Assess the operator's actions preceding and subsequent to the event. Develop a sequence of events and events causal factor analysis for the plant and operators' responses and human factors associated with the event. Compare the expected plant response to the actual plant response.

7. Assess the safety significance of the event and communicate to the Hope Creek management team the facts and safety concerns related to the problems identified.

8. Determine if any deficiencies, design vulnerabilities, etc. exist that require prompt action.

9. Assess the event with respect to previous industry and Hope Creek experiences.

10. Determine root causes and corrective actions for significant problems associated with the event.

11. Assess potential radiological consequences to personnel in the drywell during the event.

MANAGEMENT MEETING ON JULY 8, 1995 SHUTDOWN COOLING BYPASS

DISCUSSION OF EVENT

BACKGROUND ON SYSTEM

- Simplified drawing of RHR shutdown cooling system

- Parameters available to the operators
  - RHR heat exchanger inlet temperature
  - Bottom head temperature
  - Head vent temperature
  - Drywell leakage
  - Reactor Coolant System pressure

- Recirculation pump discharge valves (F031) and their design basis
  - Powered by non-safety-related power supply. No active safety related function to open or close.

- Safety related decay heat removal function is provided through vessel flood up and relief through the SRVs to the torus with torus cooling in service

MANAGEMENT MEETING ON JULY 8, 1995 SHUTDOWN COOLING BYPASS

DISCUSSION OF EVENT

SEQUENCE OF EVENTS

JULY 7, 1995

1830 Plant shutdown commenced to comply with the CREF action statement

JULY 8, 1995

0018 Plant entered Operational Condition 3

0754 "B" RHR placed in shutdown cooling

Between 0754 and 0940, both F031 valves were successfully cracked open and reclosed to prevent thermal binding

0940 Operating shift unsuccessfully attempted to stroke "A" recirculation pump discharge valve (F031A) open

0950 Operating shift unsuccessfully attempted to stroke F031A open a second time

Both attempts resulted in overloads tripping; AR was written

1057 Entered Operational Condition 4 - Cold Shutdown

1100 Cracked open the "B" recirculation pump discharge valve (F031B) and left it open to avoid thermal binding

Operators recognized core bypass flow caused by open valve but concluded it was not an issue based on temperature indication. Action not in accordance with applicable operating procedure (reactor recirc)

1152 Reactor Head Vents were opened in accordance with integrated operating procedure

1635 Removed "B" RHR from service to support surveillance test.

1709 Restored "B" RHR to service

Temperature indication for RHR HX inlet increased approximately 25F reinforcing the crew's belief that RHR was performing its intended function.

MANAGEMENT MEETING ON JULY 8, 1995 SHUTDOWN COOLING BYPASS

DISCUSSION OF EVENT

SEQUENCE OF EVENTS

1730 Operators entered drywell to assess drywell cooler coil isolation valve leak, tagout the inboard MSIVs, and investigate problem with the F031A valve

1845 F031A manually cracked open without resistance; upon exiting the drywell, the operators reported condensation on surfaces and glasses fogging while in drywell. Control room jogged F031A open electrically

1900 Shift turnover (F031A and F031B cracked open)

2000 SNSS turnover was completed; SNSS missed shift briefing.

2030 SNSS walked down the boards with the NSS and noticed "B"recirc flow at 2000 gpm, indicating bypass flow; discussed being uncomfortable with discharge valves being cracked open; decision made to close the F031A and F031E valves.

2045 Tagout of drywell primary containment instrument gas (PCIG) implemented. Cooling valves failed open providing a possible flow path to the drywell floor drain sump (DWFDS).

2100 Operators closed the F031A valve. Operators attempted to close F031B but could not; the operators jogged it open for two seconds more, and unsuccessfully attempted to close it again.

Between 2100 and 2300, operators noticed a slow increase in drywell leak detection from 0.4 gpm but attributed it to a cooling coil leak.

Chart indicated "B" recirc flow increased from 2000 to 4000 gpm (undetected).

2300 Operators noticed drywell leakage indicating between 1 to 2 gpm

MANAGEMENT MEETING ON JULY 8, 1995 SHUTDOWN COOLING BYPASS

DISCUSSION OF EVENT

SEQUENCE OF EVENTS

JULY 9, 1995

0000 Preparation commenced to support I&C testing

0100 In preparation for testing, the operators noticed a high reading on a reactor pressure trip unit and an investigation showed all four channels were between 19 and 24 psig. The readings were attributed to either elevation head or "zero" on the 1500 psig scale.

0130 Based on continuing indication of drywell leakage, the crew discussed a plan to enter the drywell to investigate and close the F031B

0230 Senior Nuclear Shift Supervisor cancelled the plan due to personnel safety concerns with footing in the drywell

0400 Crew discussed closure of the F031B when "B" RHR is removed from service based on no dp across the valve

0454 Removed "B" RHR from service to support surveillance test. The F031B was fully opened from the control room but only stroked in the closed direction for a few seconds.

When the F031B was fully opened, the charts indicate approx. 4000 gpm recirc flow; the crew expected that the valve would close with little or no dp. The crew initiated an immediate recovery plan to close the F031B after recognizing it would not close. Electrician immediately dispatched to breaker for F031B and EO dispatched to the drywell.

0500 SNSS and NSS discussed closing the "B" recirc pump suction valve (F023B) as a contingency plan if getting the F031B to close took too long. Determined no procedural guidance was available and the crew expected F031B to be closed very soon.

0508 "B" RHR was placed back in service. RHR HX inlet temperature indication increased approximately 7 F.

0550 The F031B was manually closed and shutdown cooling was fully restored with maximum RHR HX inlet temperature indicating 191 F

MANAG NT MEETING ON JULY 8, 1995 SHUTDOWN COOLING BYPASS

DISCUSSION OF EVENT

PRELIMINARY CONCERNS

Operators failed to recognize that the core bypass flow degraded the ability to remove decay heat and that temperature indications were no longer accurate.

Procedure non compliance resulted in the recirculation pump discharge valves being cracked open and left positioned off the seat. An evaluation of the effects of the valve in the maintained open position was not completed.

Management did not immediately initiate an aggressive review to assess the event.

The operating crews did not demonstrate an effective questioning attitude in reference to available indication.

The operating crews demonstrated non-conservative decision making (e.g. the F031B valve was opened fully without assurance that it could be closed).

Stroking of the recirc discharge valves is a potential operator work around that has been proceduralized and should be re-evaluated for its appropriateness as a long term solution to thermal binding.

Less than adequate follow-up on an equipment performance problem. There was no immediate troubleshooting performed on the failure of the "B" recirculation pump discharge valve (F031B) to close.

MANAGEMENT MEETING ON JULY 8, 1995 SHUTDOWN COOLING BYPASS

DISCUSSION OF EVENT

STATUS OF INVESTIGATION

STATUS OF TEAM CHARTER OBJECTIVES (Preliminary Analysis)

1. Evaluate the event for a potential mode change.

   STATUS: No mode change per Tech Spec definition, but steam pressure did develop. This determination will be validated by an independent assessment by the Nuclear Fuels Group.

2. Evaluate the event for a potential loss of shutdown cooling function. Determine if the shutdown cooling path was established in accordance with Technical Specifications and if temperature indication was adequate.

   STATUS: Shutdown cooling was degraded and did not satisfy Tech Spec Bases due to inadequate mixing.

3. Evaluate the event for potential NRC reportability.

   STATUS: Reportable under 10CFR50.73.

4. Assess the organization's, i.e., management's, response to the event.

   STATUS: Management did not immediately initiate an aggressive review to assess the event. Root cause investigation is continuing.

5. Examine any equipment failures that contributed to the event and identify root causes.

   STATUS: Recirc Discharge Valves due to thermal binding and F031B torque switch failure

6. Assess the operator's actions preceding and subsequent to the event. Develop a sequence of events and events causal factor analysis for the plant and operators' responses and human factors associated with the event.

   STATUS: Sequence of events is complete. Preliminary concerns have been developed and the root cause analysis is continuing.

MANAGEMENT MEETING ON JULY 8, 1995 SHUTDOWN COOLING BYPASS

DISCUSSION OF EVENT

STATUS OF INVESTIGATION

7. Assess the safety significance of the event and communicate to the Hope Creek management team the facts and safety concerns related to the problems identified.

   STATUS: Some preliminary concerns have been identified and are being investigated. Initial review indicates safety significance was minimal due to available coolant makeup and that equilibrium heat removal was reached.

8. Determine if any deficiencies, design vulnerabilities, etc. exist that require prompt action.

   STATUS: Procedure guidance for thermal binding of F031A(B) should be provided in the applicable operating procedure. The long term solution to the existing thermal binding issue should be re-evaluated.

9. Assess the event with respect to previous industry and Hope Creek experiences.

   STATUS: OEF data retrieval and initial review are complete. Final assessment of the event and Hope Creek responses is in progress. Previous loss of shutdown cooling events at Hope Creek are also being reviewed for similar causal factors.

10. Determine root causes and corrective actions for significant problems associated with the event.

    STATUS: The cause of the event was procedural non-compliance and failure to properly assess reactor core conditions. Corrective actions are not finalized.

11. Assess the potential radiological consequences to personnel in the Drywell during the event.

    STATUS: Preliminary ALARA data indicates an extra 6.5 mrem of a submergent dose of xenon.

- Expected review team completion: August 4, 1995

MANAGEMENT MEETING ON JULY 8, 1995 SHUTDOWN COOLING BYPASS

CORRECTIVE ACTIONS

COMPLETED ACTIONS

- July 9
  - The SNSS in charge of the shift which experienced the transient was assigned the root cause determination.
  - The acting GM assigned the Operating Engineer - Shift to review the event for any outstanding issues. (Primary issue initially identified was the maximum indicated temperature during the event of 191F with an indicated pressure of 25 psig).

- July 10
  - The acting GM directed the Operating Engineer - Staff to ensure that no other shifts take the watch believing they can interpret procedures. A night order book (NOB) entry was made requiring the SNSS's to review this event with their shifts ASAP and re-stating department expectations with regard to procedure usage
  - Engineering team established to determine if a mode change occurred and if shutdown cooling was operated in a degraded condition
  - This event was discussed at the 0930 Senior Management Issues meeting. This event was revisited on subsequent dates during this meeting.
  - The Safety Review and Quality Assurance organizations began separate independent assessments of the event.

- July 10 - 14
  - Repeated phone messages (voicemail) from the acting GM to each of the SNSSs and OEs concerning his expectations and concerns regarding the shutdown cooling event.
  - Acting GM contacted the training center to ensure this event, its root causes and corrective actions are reinforced with all shift operations personnel.

MANAGEMENT MEETING ON JULY 8, 1995 SHUTDOWN COOLING BYPASS

CORRECTIVE ACTIONS

COMPLETED ACTIONS

- July 15
  - SNSSs begin stand down meetings with each shift prior to beginning any work in the plant. Purpose of the meeting is to review effective tools to prevent errors and review these tools in the context of the recent operating events.
- July 20
  - The General Manager commissioned an independent, multi-disciplined team to evaluate the event.
- July 30
  - System operating procedures were revised based on engineering input regarding manipulation of these valves

MANAGEMENT MEETING ON JULY 8, 1995 SHUTDOWN COOLING BYPASS

CORRECTIVE ACTIONS

ACTIONS IN PROGRESS/ACTIONS TO BE TAKEN

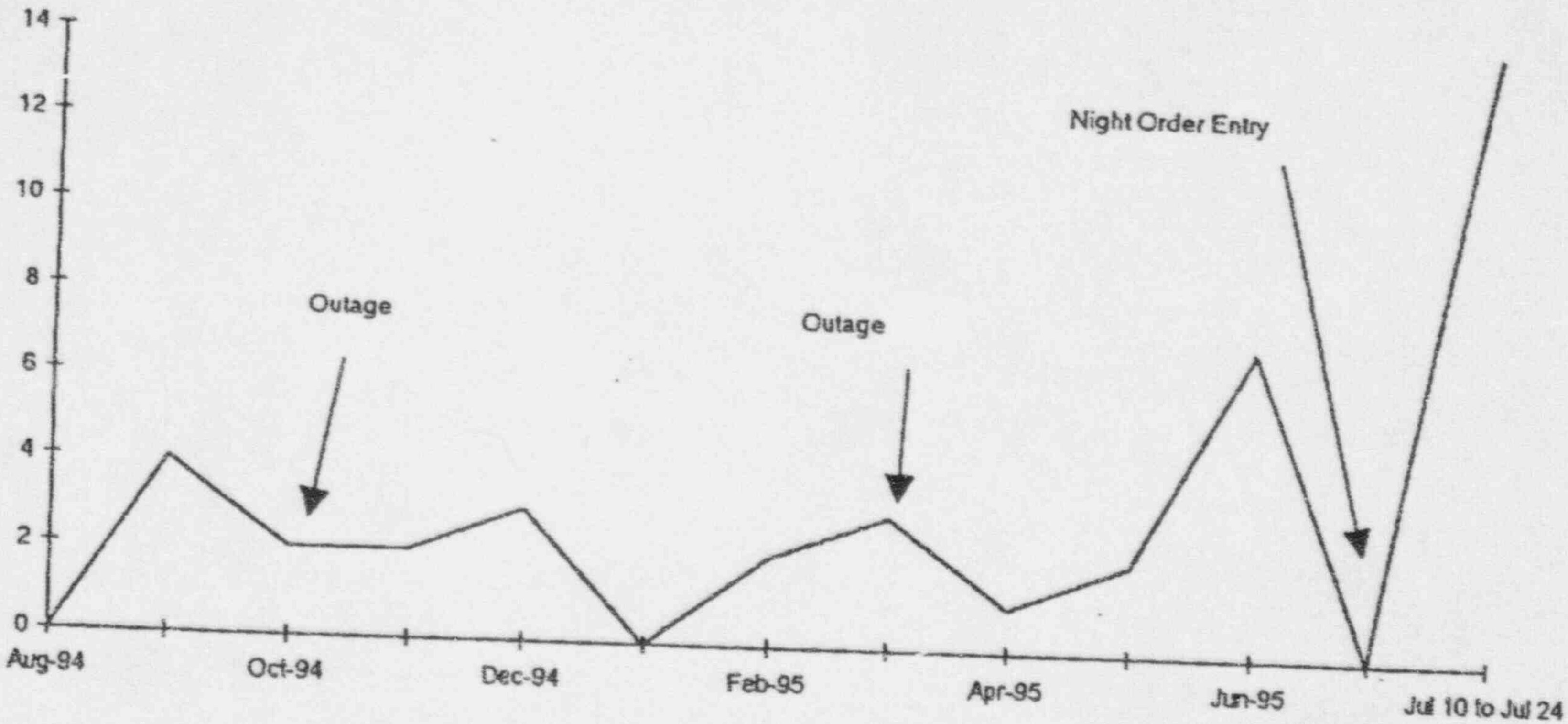
- Actions in Progress
  - Initiated a team to perform a common cause analysis to evaluate the recent increase in operator errors
  - An interim measure has been taken to assign an extra SRO to the shifts (Monday through Friday on daywork) to handle administrative burden. This action will help the crews maintain proper focus on their primary responsibilities.
- Actions to Be Taken
  - The OE - Shift will conduct a focused control room observation beginning August 7 to evaluate:
    - Procedural compliance of shift personnel
    - If control room personnel are properly focused on their roles
    - Administrative duties which should be re-assigned or eliminated

    This will allow verification of the effectiveness of our corrective actions and the findings will be reported to the Operations and General Manager
  - Cycle 1 of licensed operator requalification will include a comprehensive review of this event including root causes and corrective actions. Additionally, this event will be included in initial licensed operator training
  - Integrated and abnormal operating procedures will also be revised based on engineering input regarding manipulation of these valves
  - Additional corrective actions will be taken based upon the findings of the independent team and root cause team commissioned to address the increased operator error rate

DURING THE PERIOD AUGUST 1, 1994 TO JULY 10, 1995, WE HAD A TOTAL OF 24 OTSC's. THIS TRANSLATES TO AN OTSC GENERATION RATE OF TWO-PER-MONTH.

DURING THE PERIOD JULY 10, 1995 TO JULY 24, 1995, WE HAD A TOTAL OF 14 OTSC's. THIS TRANSLATES TO AN OTSC GENERATION RATE OF ONE-PER-DAY.

OTSC GENERATION



OPERATOR ERROR/HUMAN ERROR ANALYSIS
TEAM CHARTER

Conduct a thorough and systematic analysis of the apparent increase in operator errors during the recent forced outage. (7/8/95 - 7/25/95)

This review shall:

1. Consider the common causes among the following events:

- Condensate Pump Trip
- Shutdown Cooling Bypass
- Recirc Seal Purge Pressurization
- Extraction Steam Tagging Release with Work in Progress
- Wrong Service Traveling Water Screen Tagged

2. Assess the impact of:

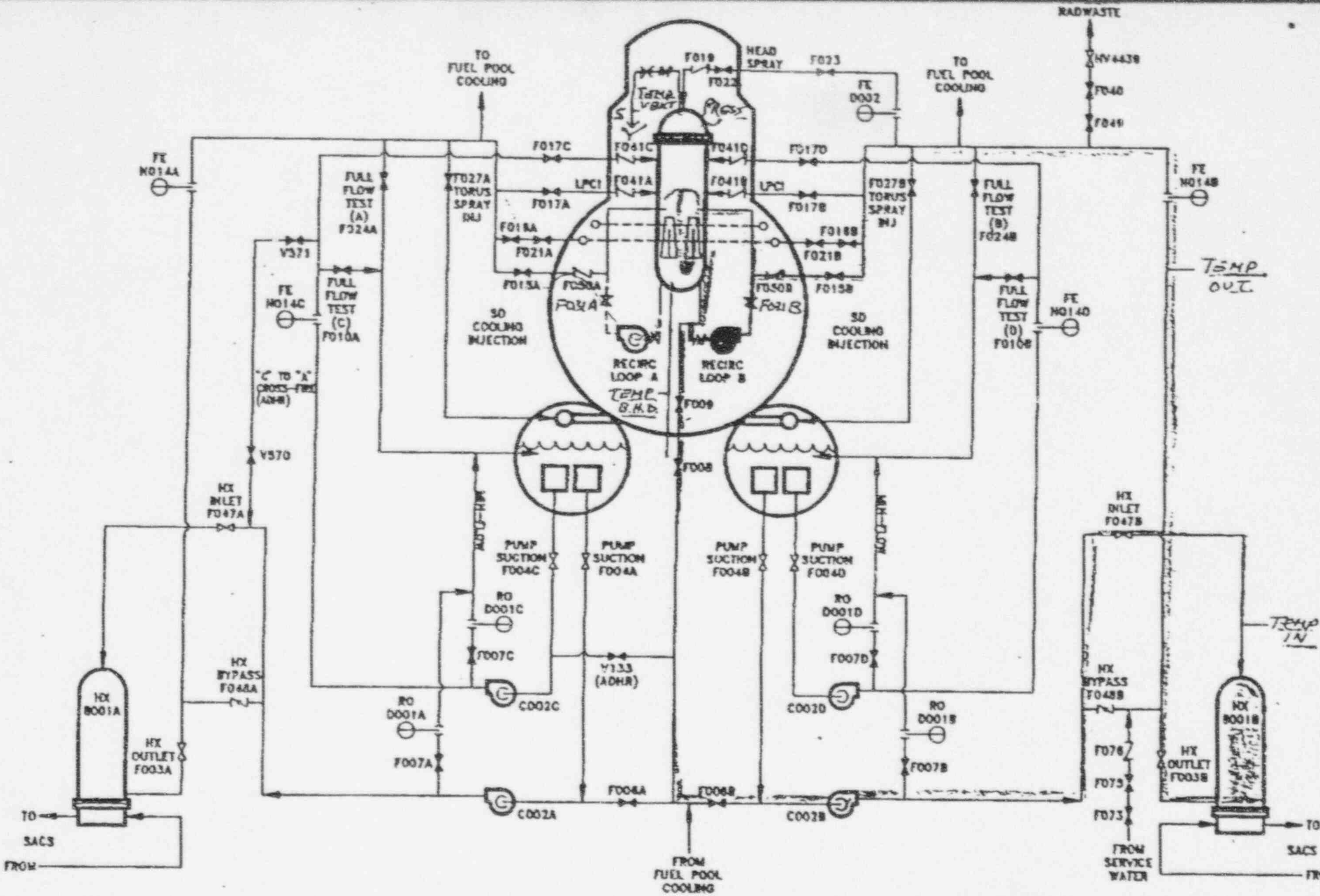
- Departmental and Organizational Changes
- Schedule Pressure
- Knowledge Deficiencies
- Distractions
- Fatigue or Illnesses
- Manning
- Task Familiarity
- Shortcuts/Procedure Violations
- Unclear Guidance
- STAR
- Role Clarity
- Communications
- Verification and Validation

3. Assess the Event with Respect to previous Hope Creek errors, particularly in the period from RF-05 to the present (inclusive)

MANAGEMENT MEETING ON JULY 8, 1995 SHUTDOWN COOLING BYPASS

CONCLUSION

- We understand the seriousness of this event and have taken or are implementing appropriate corrective actions. Any additional team recommendations will also be implemented.
- Procedural non-compliance and failure to recognize the degradation of shutdown cooling are the two significant causes
- We are deeply concerned by the recent increase in operator errors and have initiated a second root cause team to investigate this increased error rate and those errors that have occurred since the last refueling outage.
- We have reinforced our expectations for procedural adherence.
- We have placed an additional SRO on dayshift to handle administrative items. This is an interim action pending completion of an analysis of shift administrative duties by one of our Operating Engineers.
- The analysis will review the activities performed by the SNSS and NSS to identify those administrative duties which should be re-assigned or eliminated.
- Will discuss with GE and other utilities to determine the generic issues which should be communicated to the industry.
- We intend to review the necessity of preventing thermal binding of the recirc pump isolation valves while cooling down on RHR shutdown cooling.
- In the interim, our procedure has been changed to minimize the potential of RHR bypassing the core.
- Our operator training program will also be revised to include the lessons learned from this event.
- We will submit a detailed LER discussing this event, our analysis, and corrective actions.



-NORMAL FLOW

-bypass Flow

- INDICATIONS

S - STEAM CONDENSED AND Flow MEASURED

VIB	BOOK 31	LP	301H-000.00H-008C01	REV	HOPE CREEK
	PAGE 1		302H-000.00H-000028	7	
	PROG ACAD	TITLE	RHR SIMPLIFIED	FIG	
	DWG AV2415			1	

MANAGEMENT MEETING ON JULY 8, 1995 SHUTDOWN COOLING BYPASS

AGENDA

INTRODUCTION	M. Reddemann
DISCUSSION OF EVENT	C. Bauer
CORRECTIVE ACTIONS	H. Hanson
CONCLUSION	M. Reddemann

MANAGEMENT MEETING ON JULY 8, 1995 SHUTDOWN COOLING BYPASS

INTRODUCTION

- Overall Response (Event specifics to be presented later)
  - Event occurred over two shifts on July 8 and 9 during unit shutdown. Upon recognition, action requests were initiated for the recirc valve failure and the operator response to the event.
  - Due to the seriousness of the event, several other immediate actions were taken.
  - Upon return, GM formed an independent review team to conduct a thorough review of the circumstances surrounding the event.
  - Charter was developed. Team composed of individuals from:
    - Engineering
    - Training
    - Maintenance
    - Safety Review Group
    - Emergency Preparedness
    - Peer from Peach Bottom
  - Team started on July 20
  - Initiated a second root cause team to take a broad look at the recent increase in operator errors and those that occurred since the last refueling outage.
  - We will modify our operator training programs to capture the lessons learned from the event.
  - It is the GM's expectation that independent review teams be initiated immediately after a significant plant event

SHUTDOWN COOLING BYPASS EVENT REVIEW TEAM CHARTER

Conduct a thorough and systematic review of the circumstances surrounding the shutdown cooling bypass event at Hope Creek on July 8, 1995, and the resulting increase in reactor coolant temperature. The general objectives of the team are to:

1. Evaluate the event for a potential mode change.

2. Evaluate the event for a potential loss of shutdown cooling function. Determine if shutdown cooling path was established in accordance with Technical Specifications and if temperature indication was adequate.

3. Evaluate the event for potential NRC reportability.

4. Assess the organization's, i.e., management's, response to the event.

5. Examine any equipment failures that contributed to the event and identify root causes.

6. Assess the operator's actions preceding and subsequent to the event. Develop a sequence of events and events causal factor analysis for the plant and operators' responses and human factors associated with the event. Compare the expected plant response to the actual plant response.

7. Assess the safety significance of the event and communicate to the Hope Creek management team the facts and safety concerns related to the problems identified.

8. Determine if any deficiencies, design vulnerabilities, etc. exist that require prompt action.

9. Assess the event with respect to previous industry and Hope Creek experiences.

10. Determine root causes and corrective actions for significant problems associated with the event.

11. Assess potential radiological consequences to personnel in the drywell during the event.

MANAGEMENT MEETING ON JULY 8, 1995 SHUTDOWN COOLING BYPASS

DISCUSSION OF EVENT

BACKGROUND ON SYSTEM

- Simplified drawing of RHR shutdown cooling system

- Parameters available to the operators
  - RHR heat exchanger inlet temperature
  - Bottom head temperature
  - Head vent temperature
  - Drywell leakage
  - Reactor Coolant System pressure

- Recirculation pump discharge valves (F031) and their design basis
  - Powered by non-safety-related power supply. No active safety related function to open or close.

- Safety related decay heat removal function is provided through vessel flood up and relief through the SRVs to the torus with torus cooling in service