U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.:	50-293
Report No.:	50-293/90-22
Licensee:	Boston Edison Company 800 Boylston Street Boston, Massachusetts 02199
Facility:	Pilgrim Nuclear Power Station
Location:	Plymouth, Massachusetts
Dates:	October 9 - November 26, 1990
Inspectors:	J. Macdonald, Senior Resident Inspector A. Cerne, Resident Inspector W. Olsen, Resident Inspector D. Kern, Reactor Engineer

Approved by: John Rogge 12/21/90

J. Rogge, Chief, Reactor Projects Section 3A   Date

Inspection Summary:

Areas Inspected: Routine safety inspection in the areas of plant operations, radiological controls, maintenance and surveillance, safety assessment and quality verification, and engineering and technical support.

Results: Inspection results are summarized in the Executive Summary. No violations or unresolved items were identified during this inspection period.

EXECUTIVE SUMMARY

Pilgrim Inspection Report 50-293/90-22
October 9 - November 26, 1990

Plant Operations: Operators maintained effective station cognizance and control during the high pressure coolant injection system (HPCI) periods of inoperability. The October 11, 1990 response to the notification of seismic activity was well controlled and conservative.

Radiological Controls: The October 31, 1990 shipment of contaminated trash to a non-radiological waste reception center revealed several concerns regarding administrative controls in the licensee's trash compaction facility. This event was comprehensively inspected and is discussed in NRC inspection report 50-293/90-23.

Maintenance and Surveillance: Efforts to improve the HPCI system availability and reliability were noteworthy. The use of vendor expertise and data available from other industry sources, in addition to licensee resources greatly enhanced the effectiveness of the November 1, 1990 diagnostic HPCI system outage.

However, an incomplete work planning effort contributed to a partial Group II primary containment isolation system actuation (section 6.3). Additionally, a procedure revision issued in advance of a plant modification contributed to a partial reactor building isolation system actuation (Section 6.6).

Emergency Preparedness: The November 6, 1990 medical emergency drill effectively challenged the readiness of emergency response personnel.

Safety Assessment and Quality Verification: Generic NRC issues regarding potential safety-related pump loss and scram discharge volume performance were effectively addressed by licensee engineering personnel. A scram discharge volume vent and drain valve surveillance procedure concern identified by the inspector was expediently addressed by the licensee.

Engineering and Technical Support: The safety hazard evaluation conducted following identification of a reactor coolant isolation cooling system component failure was technically accurate and utilized appropriate FSAR design bases assumptions. Additionally, the voluntary submittal of a notification of valve defect to the NRC was a sound initiative.

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DETAILS

1.0 Summary of Facility Activities

Pilgrim Nuclear Power Station operated at approximately 100% power throughout the report period with the exception of a down-power evolution to approximately 60% on November 2 to backwash the main condensers. Return to 100% power was achieved on November 5, 1990.

On October 9, the licensee notified the NRC Operations Center via the Emergency Notification System (ENS) at 3:50 pm that the High Pressure Coolant Injection (HPCI) system had failed a scheduled surveillance test and had been declared inoperable (see section 6.5). Additional notifications to the NRC Operations Center via the ENS were made on October 11, at 3:38 pm to inform the NRC of a licensee press conference conducted in response to the Massachusetts Department of Public Health Leukemia Study; on October 22, at 12:22 pm to report an inadvertent actuation of the reactor building isolation system (see section 6.6); and on November 1, at 3:35 pm to inform the NRC that a bag of contaminated trash was inadvertently shipped to the SEMASS regional waste facility (see section 3.1).

On November 8 the licensee conducted an unannounced medical emergency drill to assess response personnel readiness to attend to injured and potentially contaminated plant personnel (see section 5.1).

2.0 Plant Operations (IP 71707, 71710, 92702, 90712, 93702)

2.1 Plant Operations Review

The inspector observed plant operations during regular and backshift hours in the following areas:

Control Room	Fence Line (Protected Area)
Reactor Building	Turbine Building
Diesel Generator Building	
Switchgear Rooms	

Control room instruments were observed for correlation between channels, proper functioning and conformance with Technical Specifications. Operator awareness and response to alarm conditions received in the control room were reviewed and discussed with plant operators. Operators were found cognizant of control panel and plant conditions. Control room and shift manning were in compliance with Technical Specification requirements. Posting and control of radiation, contamination and high radiation areas were inspected. Use of and compliance with radiation work permits and use of required personnel monitoring devices were verified. Plant housekeeping controls, including control of flammable and other hazardous materials, were observed. During plant tours, logs and records were reviewed to ensure compliance with station procedures, to determine if entries were correctly made and to verify correct communication of equipment status. These records included various operating logs, turnover sheets, tagout, and lifted lead and jumper logs. During routine tours of the plant, inspectors observed station security practices and noted the implementation of appropriate compensatory measures when conditions warranted. Inspections were performed on backshifts including October 9-11, 15-19, 22-25, 29 and 30, 1990 and November 5, 9, 13-16, and 21, 1990. Deep backshift inspections were performed on November 12, 1990 (a national holiday) from 8:30 am to 7:45 pm.

Pre-evolution briefings conducted in the control room were noted to be thorough with appropriate questions and answers. The operators appeared to have good knowledge of plant conditions. No unauthorized reading material was observed. Food, beverages, and hard hats were kept away from control panels.

2.2 Review of Tagging Operations

The following tagouts were reviewed with no discrepancies noted:

Tagout	Description
90-3-67	Control Rod Drive System Flow Control Valve (FCV 303-6A); Not controlling properly
90-12-24	Reactor Water Cleanup Air Operated Valve (AO-110B); Valve gasket leak
90-14-17	"A" Core Spray Pump; Resistance Testing per EQ Procedure 8.Q.3.2
90-23-41	High Pressure Coolant Injection (HPCI) System Steam Turbine; diagnostic testing and resultant maintenance activities as documented in section 4.1.
90-28-41	"A" Traveling Screen; Salt Water leak

2.3 Inoperable Equipment

Actions taken by plant personnel during periods when equipment was inoperable were reviewed to verify that technical specification (TS) limits were met, alternate surveillance testing was completed satisfactory, and equipment was properly returned to service upon completion of repairs. This review was completed for the following items:

Date Out	Date In	System
10/9	10/16	HPCI System
10/30	11/2	"A" Core Spray Pump
10/31	11/5	HPCI System
10/19	10/23	"A" Traveling Screen
11/13	11/15	Reactor Water Cleanup Pump Air Operated Valve AOV-110B
10/31	10/31	Control Rod Drive System Flow Control Valve FCV 303-6A

Control room operators maintained effective cognizance and control of plant operations during the two periods of HPCI system inoperability this inspection period. Appropriate TS limiting conditions for operation action statements were entered and required system operability verifications were completed.

2.4 Report of Seismic Activity

On October 11, the licensee was informed by Weston Observatory (via Weston Geophysical) that an earthquake had occurred earlier in the day. The earthquake epicenter was located several miles southeast of the plant and registered at 3.1 on the Richter scale. The licensee seismic recording instrumentation did not detect ground motion, nor was the event sensed by onsite personnel.

Upon notification of the earthquake, the licensee implemented seismic event response procedures which included system and structural visual inspections for physical damage, interviews of personnel onsite at the time of the event, and functional testing of the seismic recording system. No physical damage was observed and the seismic recording system was verified to have been operable.

The plant seismic recording instrumentation is an analog centralized recording magnetic tape acceleration system with remote triaxle accelerometers, peak acceleration recorders, a multichannel strong motion accelerograph, and a magnetic tape playback system. The system provides control room alarm and playback capability. The system actuates upon a ground acceleration of 0.01g. The FSAR operating base earthquake and safe shutdown earthquake ground acceleration values are 0.08g and 0.15g respectively.

The licensee response to the offsite report of seismic activity was conservative and well controlled. The inspector had no questions with respect to plant and licensee response to this event.

2.5 Operations Review Committee Composition

On August 9, 1990 the licensee submitted a proposed Technical Specification amendment to the NRC which restructured several licensee organizational units. The proposed amendment, in part removed the requirement for the Technical Section Manager to possess a current NRC issued senior reactor operator (SRO) license. This change was not significant with respect to the function of the Technical Section. However, the Technical Section Manager is also the Operations Review Committee (ORC) chairman. The inspector expressed concern that removal of this license requirement with respect to ORC functions had the potential to reduce the operational expertise of the ORC membership.

In response to this concern the licensee committed to the NRC by letter dated October 1, 1990 (BECo 90-116) to revise procedure 1.2.1, "Operations Review Committee," to require either an ORC member or alternate with a current SRO license to meet quorum requirements.

The licensee response to the inspector concern reflected a conservative safety perspective which exceeded the current licensing bases. The inspector had no further questions regarding this issue.

3.0 Radiological Controls (IP 71707)

3.1 Shipment of Contaminated Trash

On October 31, 1990, the licensee was informed by the SEMASS regional waste repository, located in Rochester, Massachusetts, that a shipment of presumed non-radiological waste alarmed the vehicle portal radiation monitors and was being returned to the licensee. Upon return to the site the vehicle was located in a secure area and a radiological survey of the shipping container contents was initiated. The survey detected two oily rags in one plastic garbage bag with a contact dose rate of 2 Mr/hour.

The licensee promptly informed the NRC resident staff of this occurrence. The resident staff notified Region I radiological protection specialists and a preliminary notification report (PN-I-90-95) was issued on November 1, 1990.

This event was inspected in detail by the regional radiological specialists during a routine inspection conducted the week of November 5, 1990. The inspection had been previously scheduled to coincide with the maintenance team inspection. The conclusions of the specialist inspection will be documented in NRC inspection report 50-293/90-23.

4.0 Maintenance and Surveillance (IP 37828, 61726, 62703, 93702)

4.1 High Pressure Coolant Injection (HPCI) System Overspeed Trips and Erratic Automatic Operation Update

During recovery from the plant transient on September 2, 1990, the HPCI system was manually initiated from the control room to aid in plant cooldown. Background information on the operational problems associated with the use of HPCI during the plant transient is described in Inspection Report 50-293/90-20. A Multi-Disciplined Analysis Team (MDAT) was formed by the licensee to investigate this transient. The MDAT analyzed the Emergency Plant Information Computer (EPIC) tracer of the event and determined that two HPCI turbine overspeed trips had occurred when the HPCI pump was started manually from the control room. These trips and other system anomalies observed on September 2 were viewed as problems requiring root cause analysis by the MDAT. The MDAT was also tasked to recommend to station management the corrective measures which needed to be implemented.

One such MDAT recommendation was that additional HPCI system surveillance be performed with instrumentation installed to verify that the HPCI overspeed trip problem had been resolved by implementation of the HPCI pump vendor suggestions and subsequent post maintenance testing. The HPCI system was declared operable on September 23, 1990. However, due to inconclusive root cause determination by the licensee relative to the HPCI system problems, this issue remained open in Inspection Report 50-293/90-20 so that NRC inspectors could continue to evaluate the effectiveness of the licensee corrective actions. During this current NRC inspection, continued troubleshooting, testing, and component replacement activities relative to the HPCI turbine have been monitored by the inspectors. The following represents a summary of the licensee activities and the results of their work up to the end of this inspection period.

On October 11, 1990, during operation of the HPCI system, the turbine was found to still have a tendency to overspeed and unacceptable flow and pressure indications were identified. Based upon troubleshooting by the licensee, it was decided to replace the electronic governor (EGR). The new EGR was found to have a different electrical polarity which necessitated a plant design change (PDC) to accomplish the required wiring revisions. After installation and calibration of the EGR, a HPCI pump and valve operability test was performed on October 15, 1990. This test indicated that the tendency for the HPCI turbine to overspeed on initial startup remained and that certain test parameters, while acceptable, were identified to have recorded test values near the maximum acceptance criteria. At this time, it was also noted that the newly installed EGR had been calibrated to settings different from the vendor's recommendation.

To address these continued HPCI turbine problems and to determine the corrective measures to be implemented during a planned system outage in November, another special MDAT was formed on October 17, 1990. A HPCI turbine vendor (Terry Turbine Co.) representative and a General Electric HPCI system engineer were requested to assist plant staff personnel with troubleshooting and repairs during the outage. Maintenance work packages were written and the work commenced on November 1, 1990 after a limiting condition for operation (LCO) was voluntarily entered and after verification that the remaining emergency core cooling systems (ECCS) were operable.

On November 5, 1990, all work on the HPCI system was completed and post work testing (PWT) commenced. During performance of the HPCI turbine speed control system calibration and testing, an unplanned overspeed trip occurred. Subsequent licensee investigation revealed that technicians had adjusted the EGR at 3800 RPM vice 3500 RPM, as required by the vendor technical manual. The higher initial turbine speed appeared to make the EGR more sensitive to the adjustment being made, causing the HPCI turbine to trip on overspeed. The licensee determined that inadequate technician training, high noise levels in the HPCI pump area, and poor communications were all contributing factors to this most recent turbine trip. The licensee immediately initiated additional on-site technician training and instruction in proper EGR adjustment techniques and verified acceptable communications had been established. Implementation of the HPCI turbine speed control system calibration procedure then continued and was completed satisfactorily on the evening of November 5, 1990. All other required PWT was performed with acceptable results.

Subsequent performance of the HPCI pump operability testing resulted in flow and pressure measurements well within the defined acceptance criteria. Additionally, the HPCI pump turbine overspeed problems were eliminated with the implementation of a pilot valve control spring modification, installed in accordance with PDC 90-65. This modification, when coupled with a realignment of the pilot relay bushing ports, resulted in the ability of the turbine speed control system to better respond to changes in demand signals. The electronic governor module (EGM) was also replaced to correct drifting problems and the ramp generator signal converter was repaired to correct damage sustained during previous troubleshooting activities.

NRC inspection of the above HPCI pump turbine repair, replacement, modification, and testing activities and review of the test results were conducted as a followup to previous inspection findings (reference: Unresolved Item 90-20-01). This item remains open and the status and condition of the HPCI system will continue to be monitored by NRC inspection to evaluate the overall effectiveness of the licensee corrective actions in this area.

5.0 Emergency Prepare.iness (IP 40500)

5.1 Medical Emergency Drill

On November 8, the licensee conducted an unannounced medical emergency drill (90-08E) to assess the readiness of emergency response personnel to attend to injured and potentially contaminated plant personnel. The drill scenario included the immediate onsite treatment of the injured worker, preparation of the injured personnel for offsite transportation, reception of offsite ambulance service, and transportation of the injured worker to Jordan Hospital.

The licensee successfully completed the drill objectives. However, during the drill the licensee identified that the station procedure to provide instruction for the transportation of injured personnel had been withdrawn. Subsequently, the fire protection staff drafted a replacement procedure which is currently in the process of review for approval.

The drill effectively challenged the licensee ability to provide effective medical treatment of a potentially contaminated injured worker, as well as ensure appropriate protective measures are implemented to minimize potential contamination or exposure to response individuals and facilities. The inspector had no unresolved questions regarding this drill.

6.0 Safety Assessment and Quality Verification (IP 35502, 92700, 92701)

6.1 NRC Bulletin 88-04

(Closed) NRC Bulletin (88-BU-04), Potential Safety-Related Pump Loss. Two pump flow design concerns were raised in Bulletin 88-04. The first concern involved potential adverse pump-to-pump interactions (e.g., dead heading) in redundant systems which share a common miniflow pipe line. The second concern involved the question of adequacy of each pump's installed miniflow capacity.

The licensee responded to NRC Bulletin 88-04 with a letter (reference BECO 88-110) dated July 13, 1988, in which the Boiling Water Reactor Owners Group (BWROG) position, as discussed in BWROG-8836, was endorsed. Plant specific information, relative to the Pilgrim pump and piping systems potentially affected by the above two concerns, was also provided in the licensee response. By letter dated July 6, 1989, the NRC Office of Nuclear Reactor Regulation (NRR) documented the review and closure of the issues discussed in Bulletin 88-04. NRR noted that the licensee had concluded that no system modifications were required and that adequate assurance was provided for fulfillment of the safety functions of the affected systems.

During this inspection, the inspector reviewed the licensee response to determine the technical justification for the assurances that safety-related system functions would not be adversely affected by redundant pump interaction or miniflow capacity. At Pilgrim the four potentially affected systems are the residual heat removal (RHR), core spray (CS), high pressure coolant injection (HPCI) and the reactor core isolation cooling (RCIC) system. In all cases, the pump minimum flow lines are orificed prior to their connection to the common full flow test line for each redundant loop. Such a design configuration eliminated any dead-heading concerns or other adverse pump-to-pump interactions.

Additionally, all four RHR pumps and both CS pumps were disassembled and inspected in 1986. NRC inspection of these maintenance activities are documented in inspection reports (IR) 50-293/86-25, 86-27, 86-34, 86-40 and 86-43. While no pump degradation was identified as a result of the minimum flow configurations, the RHR pump orifices were modified to increase minimum flows to provide additional assurance of the adequacy of pump minimum flow. With regard to the HPCI and RCIC pumps, the system design requires minimum flow only until full flow conditions are established after initiation of a HPCI or RCIC injection signal and consequent pump start. Thus, system operation under minimum flow conditions is infrequent. Also, full flow surveillance testing of both the HPCI and RCIC systems is required every operating cycle and during post-maintenance system operability checks.

In summary, based upon the orificed design piping configurations, component operating histories and maintenance and test activities for all four Pilgrim plant systems covered by the scope of Bulletin 88-04, no system modifications or operating constraints were required to correct or compensate for adverse minimum pump flow characteristics. The inspector reviewed the licensee's evaluation that system operability is not degraded by the current design and operation. No safety concerns or questions were identified and this bulletin is therefore considered closed.

6.2 SER Issue No. 41/NRC Bulletin 80-17

(Closed) Generic Safety Evaluation Report (SER) Issue No. 41; BWR Scram Discharge System (reference NRC Bulletin 80-17). Based upon NRC issuance of a generic SER in December 1980, the licensee elected to modify the Pilgrim control rod drive scram discharge volume system to eliminate the potential for undetected water in the system which could adversely affect the ability to scram. Project Design Change Request (PDCR) 82-10 was issued. A licensee safety evaluation approved the design change in March 1983, and a Confirmatory Order issued by the NRC in June 1983 endorsed the modification. The design change provided a dual volume system with redundant and diverse instrumentation.

Each of the redundant scram discharge instrument volumes (SDIV) was designed with two air operated control vent and drain valves, which fail-safe (closed) upon loss of instrument air or power to their individual operating pilot solenoid valves. Redundant level transmitters were provided to each instrument volume and corresponding control room alarms were installed, along with additional equipment failure alarms. Proposed Technical Specification changes were submitted to the NRC by BECo in June 1984 and incorporated by the NRC into Amendment No. 79 to the Operating License, issued in September 1984.

During this inspection, the inspector reviewed PDCR 82-10 and the associated safety evaluation and revised Technical Specification documentation.

The inspector examined components and instrumentation installed in the reactor building and cable spreading room, as such equipment related to the subject SDIV design change. The inspector also checked and discussed with operations personnel the alarms and trip signals on Panel C905 which are generated by the redundant SDIV instrumentation. Since installation of the reactor protection system analog trip system (RPS ATS) cabinets occurred subsequent to the implementation of PDCR 82-10, the inspector compared the field installed SDIV level element/transmitter tagging with the RPS ATS circuitry scram rod block signal nomenclature to determine whether trip system/channel logic consistent with the instrument diagrams had been maintained. It was noted that new control room Panel C905 alarm windows associated with RPS ATS test or power supply failures were installed to replace the PDCR 82-10 equipment failure alarms.

Additionally, the inspector reviewed the Pilgrim Technical Specifications (i.e., section 3/4.3.G) governing the scram discharge volume and examined the BECo procedures for the SDIV vent and drain valve testing and timing intended to comply with the technical surveillance requirements. During the review of PNPS procedure 8.3.3, "Scram Discharge Instrument Volume Vent and Drain Valve Quarterly Operability," it was noted that quarterly stroking of each set of redundant SDIV vent and drain valves is accomplished by exercising the air dump system test switch on control room panel C905. Placement of the test switch in the "ISOLATE" mode closes the vent and drain valves. Such test switch selection causes energization of the SDIV test solenoid valve, the consequent venting of instrument air and resulting fail-safe closure of the vent and drain valves. Returning the air dump system test switch to "NORMAL" opens the same SDIV vent and drain valves.

With respect to PNPS Procedure 8.M.1-31, "SDV Vent and Drain Timing," the air dump system test switch is again used in the reactor scram reset sequence of operations following SDIV vent and drain valve timing to comply with the surveillance requirements of technical specification 4.3.G.2.a. However, the inspector questioned whether the sequence of operations specified in procedure

8.M.1-31 (i.e., test switch "ISOLATE" scram "RESET," test switch "NORMAL") met the full intent of Technical Specification 4.3.G.2.b intended to verify that the valves open when the scram is reset. Since by design, the valves open upon placing the test switch in "NORMAL," the role of the scram reset signal in allowing the valves to open could be questioned. Contributing to this question is the fact that the procedural requirements for operator actions for reset subsequent to actual reactor scrams, as specified in PNPS procedure 2.1.6, "Reactor Scram," delineate a sequence of operational steps (i.e., scram "RESET," test switch "ISOLATE," test switch "NORMAL") different than the procedure 8.M.1-31 sequence.

The inspector discussed the above question with the licensee Instrumentation and Control department supervisory personnel. It was agreed that adding steps to the current procedure 8.M.1-31 sequence of operations would adequately address this question and fully satisfy the intent of Technical Specification 4.3.G.2.b. A procedure change notice (PCN) to procedure 8.M.1-31 was initiated to effect a revision which checks that the vent and drain valves will not open upon placing the test switch in "NORMAL" until the scram is "RESET." By adding these procedural steps prior to resetting the scram signal the surveillance requirement to ensure that the SDIV valves "open when the scram is reset" is more fully and verifiably achieved.

The inspector had no further questions on the scram discharge volume surveillance requirements as they related to the PDCR 82-10 modifications. The Master Surveillance Tracking Program was examined to verify that the Technical Specification 4.3.G surveillance commitments were properly scheduled. No problems were identified with either the current operability of the scram discharge volume system or the implementation of the PDCR 82-10 modifications intended to address the NRC Generic SER and Bulletin 80-17 concerns.

This issue is considered closed.

6.3 <u>LER 90-15</u>

LER 90-15, "Unplanned Partial Isolations of the Hydrogen and Oxygen (H2O2) Analyzer System and the Reactor Coolant Pressure Boundary Leak Detection System During Jumper Installation," addressed the September 13, 1990 partial primary containment isolation system (PCIS) Group II actuation. The event occurred while the plant was shutdown. The actuation was experienced during the installation of a temporary ground jumper lead necessary to facilitate the replacement of an electrical relay in the H2O2 analyzer system. The temporary jumper lead inadvertently contacted two terminal lands adjacent to the intended land terminal point. This caused the permanent electrical connections to ground, the associated fuses to blow, and the associated relays to de-energize. Deenergization of the affected relays caused the H2O2 and the reactor coolant pressure boundary leakage system isolation valves powered by the affected relays to essentially receive isolation signals and to close or remain closed.

Following the partial isolation, the blown fuses were replaced and the affected valves were returned to normal configurations. The licensee conducted a critique of the event which determined that the event occurred as a result of inadequate work planning. The relay on which the temporary ground was to be landed is installed in a location which is difficult to access. Additionally the relay terminal lands are in very close proximity to each other, making it extremely difficult to identify and contact only the desired terminal land. The work plan was revised to install the temporary ground to the proper terminal land, with visual and electrical verification before connecting the temporary lead to ground.

Long-term licensee corrective actions included the formation of a task force to identify potential problems and propose enhancements when implementing the lifted lead and jumper program.

This LER effectively addressed reporting criteria, including similar previous events. The event occurred with the plant in shutdown and was of minimal safety significance. The inspector had no additional questions regarding this LER.

6.4 LER 90-16

LER 90-16, "Automatic Closing of the Group I Isolation Valves While Shutdown due to High Reactor Water Level," addressed the September 17, 1990 automatic closure of the main steam isolation valves (MSIVs) when the shutdown cooling system (SDC) was secured in preparation for reactor startup. This event was documented in NRC inspection report 50-293/90-20, section 2.8.

The MSIVs isolated when the PCIS Group I high reactor vessel water level was reached following the securing of the SDC system. The high vessel water level was the result of: an initially high vessel water level before SDC was secured, closure of the RHR discharge valve in very close order (approximately six seconds), and a higher than normal reactor coolant temperature. Following the isolation, operators re-initiated the SDC system, lowered vessel water level, reset the Group I isolation and reopened the MSIVs. The event was adequately reviewed, the SDC system was secured satisfactorily, and plant startup preparations were resumed.

The licensee drafted a revision to the RHR procedure to establish a reactor vessel water level operating band to accommodate RHR pump starts and stops. The revision also addresses two (RHR) pump operation while in the SDC mode, and improves the instruction for securing the SDC system.

The event occurred while the plant was shutdown and was of minimal safety significance. The inspector had no additional questions regarding this LER.

6.5 LER 90-17

LER 90-17, "High Pressure Coolant Injection System Declared Inoperable Due to Overspeed During Surveillance Testing," addressed the October 9, 1990 HPCI system overspeed trip. Following the HPCI system trip the system was declared inoperable and a seven day TS limiting condition for operation was entered.

The LER appropriately addressed the reporting criteria. Additionally the report provided effective development and discussion of the recent HPCI system anomalies. The inspectors identified an unresolved item in NRC inspection report 50-293/90-20 (90-20-01) regarding the HPCI system. Section 4.1 of this report serves to update the unresolved item, as well as address this event and the subsequent HPCI maintenance outage. The inspector had no additional questions regarding this LER.

6.6 LER 90-18

LER 90-18, "Inadvertent Actuation of a Portion of the Secondary Containment System During Surveillance Testing Due to Procedure Error," addressed the October 22, 1990 actuation of the "A" train of the reactor building isolation system (RBIS) while performing a semi-annual surveillance functional test. The actuation caused closure of the "A" train reactor building supply and exhaust ventilation dampers and start of the "A" train of the standby gas treatment system. The isolation occurred while operators were attempting to restore the RBIS after aborting a logic system functional test when licensee personnel determined the test procedure instruction was incorrect.

Procedure 8.M.2-1.5.8.1, revision 16, "High Drywell Pressure, Low Water Level and High Radiation Logic System A - Inboard Functional Test," was recently issued (July 14, 1990) to reflect a system modification scheduled to be completed July 17, 1990. However, the modification was postponed until the 1991 refueling outage when the potential for adverse operational impact would be minimized. Although the modification was postponed, the revised procedure was not restored to its previous revision instruction.

While performing the procedure, technicians noted the procedure (and hardware) discrepancy. The procedure was aborted and technicians attempted to restore the RBIS normal configuration. When the system logic switch was repositioned from the test logic to the standby position, the control and seal-in circuit remained de-energized and an isolation signal was generated. Following the event the licensee restored the logic and reset the isolation.

The cause of the event was determined by the licensee to be procedural error, in that procedure 8.M.2-1.5.8.1 was revised and issued in advance of implementation of the modification. The licensee upgraded interdepartmental controls of modifications. Additionally, the licensee is developing guidance for system restorations if an activity cannot be completed. The inspector had no additional questions regarding this LER.

7.0 Engineering and Technical Support (IP 40500)

7.1 Licensee Notification of Valve Defect

On September 2, 1990, during recovery from a manual scram, the RCIC suction piping was momentarily pressurized. Licensee investigation of the pressurization determined a design defect prevented the RCIC discharge check valve from fully seating following a RCIC turbine trip which allowed high pressure reactor water cleanup system backflow through the RCIC system. This event is documented in NRC inspection report 50-293/90-20, section 6.1.

Following correction of the design deficiency the licensee initiated a safety hazards evaluation to determine if the deficiency was reportable to the NRC consistent with the criteria of 10 CFR 21. The evaluation which was completed October 18, 1990, concluded the deficient check valve design did not pose a significant hazard to the health and safety of the public with respect to the RCIC system or in any other application at PNPS. The evaluation referenced appropriate FSAR bases and was adequately developed from a technical perspective.

Although the evaluation concluded this condition was not safety significant at PNPS, the licensee voluntarily submitted a letter to the NRC reporting a notification of valve defect. The letter dated November 14, 1990 effectively developed the event scenario and referenced the associated LER (90-13). The letter also detailed the failure mechanism due to valve design as well as the vendor approved modification.

The licensee evaluation was technically sound, timely and appropriately addressed the safety significant hazard criteria of 10 CFR 21. Additionally, the licensee demonstrated a safety awareness commensurate with the potential generic application of this problem by the voluntary submittal of the design-defect letter.

The inspector had no further questions regarding this issue.

8.0 NRC Management Meetings and Other Activities (IP 30703)

8.1 Routine Meetings

At periodic intervals during this inspection, meetings were held with senior plant management to discuss licensee activities and areas of concern to the inspectors. On December 18, 1990, the resident inspector staff conducted an exit meeting with BECo management summarizing inspection activity and findings for this report period. No proprietary information was identified as being included in the report.

8.2 Other NRC Activities

A Region I radiological controls inspection was conducted November 5-9, 1990 (Inspection Report 50-293/90-23).

An NRC Region I maintenance team inspection (MTI) was conducted November 5-16, 1990 (Inspection Report 50-293/90-80). The Region I Reactor Projects Branch Chief and the NRR Licensing Project Manager responsible for PNPS were onsite November 16, 1990 to attend the initial MTI exit meeting.