UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

PDR

CHAIRMAN

February 7, 1991

Dr. Donald M. Muirhead, Jr., and
Mrs. Mary C. Ott,
Co-Chairmen, Duxbury
Citizens Urging Responsible Energy
P.O. Box 2621
Duxbury, Massachusetts 02331

Dear Dr. Muirhead and Mrs. Ott:

I am responding to your letter of December 5, 1990, in which you expressed your concerns regarding the September 2, 1990 manual scram and subsequent plant shutdown at Pilgrim Nuclear Power Station (PNPS). The Nuclear Regulatory Commission (NRC) staff reviewed this event and documented its findings in NRC Inspection Report 50-293/90-20 dated November 7, 1990. I have enclosed a copy of the report for your information. (Enclosure 1).

With respect to event classification, the staff concluded that the Boston Edison Company's (the licensee's) response to the event was proper, including its evaluation of the emergency action level classification and determination not to activate the emergency plan. (See Enclosure 1, page 16). The licensee did notify or attempt to notify Commonwealth and town officials when it implemented procedure EP-AD-130, "Responsibilities of On Call Management Representatives," Revision 2, which is the appropriate action for events significant enough to warrant increased awareness of plant management but not serious enough to require implementation of the emergency plan. The NRC was notified of the event as required by 10 CFR 50.72. The staff reviewed this event report and concluded that the event did not involve potential degradation of the level of safety of the plant and did not require classification within an emergency class in accordance with the PNPS classification procedure.

You cited NUREG-0654 as the basis for your belief that an "Unusual Event" should have been declared in response to this event. The staff's response to your specific comments is provided in Enclosure 2. I want to emphasize, however, that NUREG-0654, Appendix 1, provides guidance in the form of example initiating conditions for developing emergency action levels (EALs). Licensees use this guidance to develop plant specific EALs, which may differ from the example conditions in NUREG-0654 depending on individual plant systems and operating procedures. The Pilgrim EALs have been reviewed by the NRC and have been found to adhere to regulatory requirements and to meet the intent of the guidance of NUREG-0654. The NRC is in the process of conducting a generic review of the EAL guidance contained in NUREG-0654. The Pilgrim event will be one of the scenarios examined to determine if additional guidance is warranted.

9102120416 XA

910221

DF02
111

In response to your concern that the operator's action to manually scram the reactor was based on the economic incentive stated in the licensee's rate case agreement with the Department of Public Utilities, the staff concluded that the licensee acted appropriately. The control room operators used appropriate procedures, evaluated valid parameter data, and correctly determined that the event resulted from a problem in the feedwater control system which caused the water level in the reactor vessel to increase slowly. Once the operators determined that the actions they took to correct the feedwater control system malfunction were unsuccessful, they promptly initiated a manual reactor scram.

You also requested a transcript of the September 12, 1990 meeting between the NRC and the licensee in our Region I office. Although the meeting was not transcribed, Section 7.3 of the enclosed inspection report documents the substance of conference calls and meetings in response to the September 2, 1990 event, including the September 12, 1990 meeting.

The Commission believes that the licensee and the NRC staff responded appropriately to this event.

Sincerely,

Kenneth M. Carr

Kenneth M. Carr

Enclosures:
1. IP 50-293/90-20
2. Staff Analysis

Enclosure 1

NOV 07 1990

Docket No. 50-293

Boston Edison Company
ATTN: Mr. Ralph G. Bird
Senior Vice President-Nuclear
Rocky Hill Road
Plymouth, Massachusetts 02360

Gentlemen:

Subject: NRC Region I Inspection Report No. 50-293/90-20

This refers to the routine safety inspection conducted by Messrs. J. Macdonald, A. Cerne, W. Olsen, and D. Kern of this office August 16 - October 8, 1990 at the Pilgrim Nuclear Power Station, Plymouth, Massachusetts. Areas examined during this inspection are described in the NRC Region I Inspection Report which is enclosed with this letter.

The September 2 manual reactor scram and plant shutdown was extensively reviewed by the resident staff in this report as well as by the Special Inspection Team dispatched to the site in response to the event (Inspection Report 50-293/90-21). Region I management concluded that the licensed operators on shift at the time of the event displayed strong performance in response to the component failures and system malfunctions encountered while bringing the plant to cold shutdown conditions. Additionally, the multi-disciplinary analysis team (MDAT) provided effective and comprehensive investigation of each event anomaly and recommended appropriate specific and programmatic corrective actions.

Notwithstanding effective event response, we remain concerned with the number of challenges presented to the operators during the event and the apparent contribution of the maintenance program to these challenges. We are aware of your self assessments of the maintenance program as well as the enhancements planned and implemented in response to weaknesses you identified. We will assess the effectiveness of these changes and the maintenance program in general during the NRC Maintenance Team Inspection to be conducted November 5-16, 1990.

Additionally, it appears that one of your activities involving the temporary leak repair of a shutdown cooling system suction valve was not conducted in accordance with NRC requirements. You are required to respond to this letter following the instructions in the enclosed Notice of Violation. In your response you should address the adequacy of the temporary leak seal program in general and specifically describe administrative measures in place to ensure that quality standards are maintained during program application.

IE01

Your cooperation with us is appreciated.

Sincerely,

Original Signed By:
Jon R. Johnson

Jon R. Johnson, Chief
Projects Branch No. 3
Division of Reactor Projects

Enclosures:
1. Notice of Violation
2. NRC Region I Inspection Report No. 50-293/90-20

cc w/encls:
R. Anderson, Vice President, Nuclear Operations and Station Director
E. Kraft, Plant Manager
J. Dietrich, Licensing Division Manager
R. Swanson, Regulatory Affairs Manager
E. Robinson, Nuclear Information Manager
R. Hallisey, Department of Public Health, Commonwealth of Massachusetts
R. Adams, Department of Labor and Industries, Commonwealth of Massachusetts
D. Long, Security Group Leader
The Honorable Edward M. Kennedy
The Honorable John F. Kerry
The Honorable Edward J. Markey
The Honorable Edward P. Kirby
The Honorable Peter V. Forman
The Honorable Nicholas J. Costello
The Honorable Lawrence R. Alexander
B. McIntyre, Chairman, Department of Public Utilities
Chairman, Plymouth Board of Selectmen
Chairman, Duxbury Board of Selectmen
Plymouth Civil Defense Director
P. Gromer, Massachusetts Secretary of Energy Resources
Sarah Woodhouse, Legislative Assistant
A. Nogee, MASSPIRG
Regional Administrator, FEMA
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
NRC Resident Inspector
Commonwealth of Massachusetts SLO Designee

bcc w/Executive Summary Only:
B. Hehl, RI
J. Wiggins, RI
M. Knapp, RI
W. Hodges, RI
J. Durr, RI
L. Betttenhausen, RI

bcc w/encls:
Region I Docket Room (with concurrences)
Management Assistant, DRMA (w/o encls)
J. Johnson, DRP
J. Rogge, DRP
J. Macdonald, SRI - Pilgrim (w/concurrences)
K. Abraham, PAO (2) All Inspection Reports
J. Caldwell, EDO
R. Wessman, NRR
R. Eaton, NRR
N. Conicella, RI

RI:DRP	RI:DRP	RI:DRP
Macdonald/meo	Rogge	Johnson
11/7/90	11/7/90	11/7/90

OFFICIAL RECORD COPY

A:PIL9020.MEO

NOTICE OF VIOLATION

Boston Edison Company
Pilgrim Nuclear Power Station

Docket No. 50-293
License No. DRP-35

As a result of the inspection conducted on August 16 to October 8, 1990 and in accordance with the NRC Enforcement Policy (10 CFR 2, Appendix C), the following violation was identified:

> 10 CFR 50, Appendix B, Criterion III requires that design changes, including field changes, be subject to design control measures commensurate with those applied to the original design and be approved by the organization that performed the original design.
>
> The Boston Edison Quality Assurance Manual, Volume II, Section 3 requires that, subsequent changes to an approved plant design change package are made using Field Revision Notices (FRNs). Nuclear Engineering Department Procedure (NED), 3.03, "Field Revision Notices," states that: The cognizant discipline division manager (DM) or an engineer assigned by the DM review FRNs and make determinations to whether they are major or minor and safety related or non-safety related. A major FRN is defined as one that: (1) changes the original conceptual design or the intent of the implementation document, or (2) affects the (bases) on which the approved safety evaluation was made. NED 3.03 also requires that the original design change safety evaluation be revised (to reflect the affect of major FRNs) and approved prior to implementation.
>
> Contrary to the above, on September 11 and September 23, the licensee issued two minor FRNs which affected the safety evaluation bases of a plant design change for the temporary leak seal repair of a shutdown cooling system valve. As a result, the appropriate safety evaluation (revision) and required approval did not occur.
>
> This is a Severity Level IV Violation (Supplement I).

Pursuant to the provisions of 10 CFR 2.201, Boston Edison Company is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Regional Administrator, Region I, and a copy to the NRC Resident Inspector, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps which will be taken to avoid further violations, and (4) the date when full compliance will be achieved. If an adequate reply is not received within the time specified in this Notice, an order may be issued to show cause why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

9011140171

The response directed by this Notice is not subject to the clearance procedures of the Office of Management and budget as required by the paperwork Reduction Act of 1980, PL 96-511.

U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.: 50-293

Report No.: 50-293/90-20

Licensee: Boston Edison Company
800 Boylston Street
Boston, Massachusetts 02199

Facility: Pilgrim Nuclear Power Station

Location: Plymouth, Massachusetts

Dates: August 16 - October 8, 1990

Inspectors: J. Macdonald, Senior Resident Inspector
A. Cerne, Resident Inspector
W. Olsen, Resident Inspector
D. Kern, Reactor Engineer

Approved by: John Rogge 11/7/90
J. Rogge, Chief, Reactor Projects Section 3A Date

Inspection Summary: Inspection on August 16 - October 8, 1990 (Report No. 50-293/90-20)

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

Results: Inspection results are summarized in the attached Executive Summary. One violation in the area of engineering and technical support was identified for failure to perform appropriate evaluation of two revisions to a design change for the temporary leak seal repair of a shutdown cooling suction valve.

EXECUTIVE SUMMARY

1. Plant Operations: Operators displayed excellent transient response knowledge during the September 2 manual reactor scram. Appropriate procedures were utilized and the plant was maintained in a safe condition throughout the event. The operators also effectively ensured positive control over all activities during extended plant startup testing.

   Maintenance and Surveillance: The component failures and system malfunctions which presented operational challenges during the September 2 event were partially attributed to inadequate maintenance program implementation. Although the plant was maintained in a safe condition, the diverse equipment complications cause NRC concern.

   Emergency Preparedness: The September 2 event was appropriately reviewed with respect to Emergency Plan emergency action level criteria. Proper state and local notifications were completed in accordance with administrative procedures.

   Safety Assessment and Quality Verification: The multi-disciplinary analysis team (MDAT)investigation of the September 2 event was effective and well focused. The MDAT was fully supported by senior management. The MDAT report and subsequent operations review committee event review demonstrated continued improvement in the licensee self identification, assessment, and corrective actions capabilities.

   Engineering and Technical Support: The engineering analysis of the reactor core isolation cooling system suction line pressurization event was comprehensive and utilized conservative assumptions. However, engineering support and disposition of revisions to a design change for the temporary leak seal repair of a shutdown cooling suction valve was inadequate and resulted in the design change safety evaluation bases being adversely impacted.

2. Violation:

   One violation was identified as a result of the failure of the licensee to perform appropriate evaluation of two revisions to a design change for the temporary leak seal repair of a shutdown cooling suction valve.

3. Unresolved item:

   One unresolved item was identified to review and assess results of the increased high pressure coolant injection system surveillance testing periodicity and data acquisition capability.

TABLE OF CONTENTS

Table of Contents

*The NRC Inspection Manual inspection procedure (IP) that was used as inspection guidance.

DETAILS

1.0 Summary of Facility Activities

Pilgrim Nuclear Power Station was at 100% power at the beginning of the report period. On August 29, 1990 a previously identified minor packing leak on the "B" feedwater regulating valve (FRV) increased significantly. On August 30, reactor power was reduced to approximately 46% to facilitate repair of the "B" FRV and to accomplish a backwash of the main condenser. Reactor power returned to 100% on September 1.

On September 2, at 10:33 p.m. a manual reactor scram was initiated due to an increasing reactor vessel water level caused by a component failure in the feedwater regulating valve control air system. The shutdown following the reactor scram was complicated by several component failures and system malfunctions which are discussed in detail in this report. The licensee entered a fifteen day forced outage after plant shutdown to investigate equipment challenges during the event and to implement corrective measures.

On September 17, plant startup was initiated. The reactor was maintained at approximately 120 psi and 1% of rated power for several days to facilitate post maintenance and operability testing of steam turbine driven core cooling systems and to perform a temporary leak sealing technique on a shutdown cooling system valve.

Following completion of these activities power ascension was commenced. The turbine generator was synchronized to the offsite distribution system on September 25. The reactor attained 100% of rated power on September 28. At the conclusion of the inspection period, the plant was operating at 100% power.

On September 3, the licensee notified the NRC Operations Center via the Emergency Notification System (ENS) at 1:02 a.m. to report the manual reactor scram and to report that the reactor core isolation cooling system had malfunctioned following the scram and had been declared inoperable. Additional notifications to the NRC Operations Center via the ENS were made on September 3, at 3:17 a.m. to report automatic Group I, Group II and Group VI primary containment isolation system (PCIS) actuations during plant shutdown and at 4:38 p.m. to report an automatic Group III PCIS actuation following the initiation of the shutdown cooling system. These notifications were made in accordance with the requirements of 10 CFR 50.72. Notification via ENS to the NRC was also made on September 13 to report a partial PCIS actuation of portions of the primary atmospheric sample system during a maintenance activity and on September 17 to report an automatic partial Group I PCIS actuation which occurred when the shutdown cooling system was secured.

On September 27, the licensee announced several management changes. Effective December 1, 1990, Mr. Stephen J. Sweeney will be retiring as the BECo Chief Executive Officer (CEO) but will remain Chairman of the Board of Directors. Mr. Bernard W. Reznicek, currently the BECo President and Chief Operating Officer, will succeed Mr. Sweeney as CEO. Also effective December 1, 1990, Mr. Ralph G. Bird, currently the BECo Senior Vice President of Nuclear Operations (SVP-N), will assume an Executive Vice President position responsible for all engineering and production operations. Mr. George Davis, currently the BECo Vice President of Nuclear Administration, will succeed Mr. Bird as Senior VP-N.

Additionally, the licensee announced that effective September 27, Mr. Roy Anderson, current Plant Manager, was selected to succeed Mr. Kenneth Highfill as Vice President of Nuclear Operations and Station Director. Mr. Edward S. Kraft, current Deputy Plant Manager and Acting Plant Manager, was selected to succeed Mr. Anderson as Plant Manager.

On October 3, the NRC Region I Regional Administrator was onsite to meet with the resident inspectors, to tour the facility, and meet briefly with licensee management.

On September 10, the NRC Region I Director of Reactor Projects was onsite to meet with the resident inspectors and to tour the facility.

An NRC Region I Special Inspection was conducted in response to the September 2-3 manual reactor scram and shutdown to; evaluate licensee performance during the event; evaluate the effectiveness of the licensee investigation of the event; and evaluate maintenance program contribution to the event. The special inspection was conducted on September 5-7 (Inspection Report 50-293/90-21).

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

2.1 Plant Operations Review

The inspectors observed plant operations during regular and backshift hours of the following areas:

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

Control room instruments were observed for correlation between channels, proper functioning and conformance with Technical Specifications. Alarms received in the control room were reviewed and discussed with the operators. Operator awareness and response to these conditions were reviewed. Operators were found cognizant of board and plant conditions. Control room and shift manning were compared 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 checked. 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 the lifted lead and jumper logs. Inspections were performed on backshifts including August 15, 21-24, and 27-31, September 4-6, 10, 17-20, 24 and 25, October 1, and 3-4, 1990. Deep backshift inspections were performed as follows:

Date	Time
September 2	2:30 pm - 9:45 pm
September 3	8:30 am - 2:15 pm
September 5	3:30 am - 5:00 am

Pre-evolution briefings 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 Safety System Reviews

Portions of the emergency diesel generators, reactor core isolation cooling, high pressure coolant injection, residual heat removal and safety related electrical systems were reviewed to verify proper alignment and operational status in the standby mode. The review included verification that accessible major flow path valves were correctly positioned, power supplies were energized, lubrication and component cooling water was proper, and components were operable based on a visual inspection of equipment for leakage and general conditions. No violations or safety concerns were identified.

2.3 Review of Tagging Operations

The following tagouts were reviewed with no discrepancies noted:

Tagout	Description
90-6-42	Startup Feed Regulating Valve; Perform air tightness test on AO-643
90-6-43	Feedwater Regulating Valve AO-642A; Implementation of PDC-90-56
90-6-3?	Feedwater Regulating Valve AO-642B; Investigate cause of failure of the valve
90-13-26	Reactor Core Isolation Cooling System; Repair pressure indicator PI-1360-20
90-23-33	High Pressure Coolant Injection System; Investigate cause of turbine trip
90-10-54	Shutdown Cooling Suction Valve MO-1001-50; Investigate cause of valve not stroking open (on September 3)

2.4 Inoperable Equipment

Actions taken by plant personnel during periods when equipment was inoperable were reviewed to verify that technical specification 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
9/2	9/24	Reactor Core Isolation Cooling System
9/2	9/24	Feedwater Regulating Valves A&B
9/2	9/25	Startup Feed Water Regulating Valve
9/2	9/24	High Pressure Coolant Injection System
9/2	ROOS*	Shutdown Cooling Suction Isolation Valve (MO-1001-50)

*ROOS - remained out of service. In accordance with TS 3.7.A, downstream isolation valve MO-1001-47 was administratively controlled in the closed position.

2.5 Operational Safety Findings

Licensee administrative control of off-normal system configurations by use of temporary modifications and tagging procedures was in compliance with procedural instructions and was consistent with plant safety. Overall plant cleanliness and material condition continued to be good.

2.6 Manual Reactor Scram

On September 2, 1990 at 10:33 p.m. the licensee initiated a manual reactor scram from approximately 100% power due to increasing reactor vessel water level. Approximately a half hour before the scram, operators began to receive high reactor vessel water level alarms and experienced difficulty with feed water flow control. Operators successfully stabilized vessel level by cycling feed pump recirculating valves. However, it became apparent that the main feed regulating valves were not responding properly to signal demands and the operators initiated a reactor scram.

Following the scram, operators attempted to initiate reactor core isolation cooling (RCIC) to provide vessel water level control. The RCIC turbine immediately tripped on overspeed; it was restarted twice, tripped again both times, and remained unavailable. Operators utilized the feedwater start-up feed regulating valve (8-10% capacity) and a single feed pump to control vessel water level. This valve operated erratically and appeared to be fully open, providing excessive water to the vessel. The inability to gain fine vessel water level control resulted in a Group I isolation (Main Steam Isolation Valve (MSIV) closure) on high reactor vessel water level. Operators manually initiated the high pressure coolant injection (HPCI) system to provide reactor vessel water level control. However, HPCI was designed to deliver approximately 4000 gallons per minute to the reactor vessel in the event of a loss of coolant scenario and also provided excessive water to the vessel.

Reactor pressure control was accomplished by cycling safety-relief valves and by operation of HPCI in full flow test. At approximately 2:00 a.m. on September 3, the MSIVs were reopened. The plant was then cooled down and depressurized normally.

At approximately 1:00 p.m. on September 3, while attempting to establish shutdown cooling system (SDC) flow, SDC suction valve SDC-50 did not stroke full open. Troubleshooting revealed a failed seal-in relay. The valve was opened and SDC was initiated at 4:4. p.m. Immediately following initiation, SDC automatically isolated due to a pressure transient. After SDC system venting and inspection, SDC was successfully established at 5:33 p.m. A complete chronology and assessment of operator actions and plant responses to this event is documented in Inspection Report 50-293/90-21.

The inspectors, in conjunction with the special inspection team, determined that the operating staff performed appropriately in mitigating the operational challenges presented during this event.

2.7 Plant Procedure Revisions as a Result of the September 2, 1990 Event

Review of this event, as documented in Special Inspection Report 50-293/90-21, identified several inadequate procedures which did not provide sufficient instruction to operators to mitigate an event involving increasing reactor water level. Additionally, the Special Inspection Report concluded that the instruction of several procedures was incorrect and contributed to system malfunctions during the September 2 event. A summary of each procedure revision by the licensee in response to this event is provided below. The revisions were reviewed by Special Inspection team members as well as the resident staff and were determined to be appropriate.

2.7.1 Feedwater Regulating Valves

Plant procedure 2.4.49, "Loss of Normal Feed and Feedwater Control Valve Malfunction," was revised to include instruction to position reactor feed pump recirculation flow valves to control reactor vessel water level in the event of a feedwater regulating valve failure resulting in an increasing reactor vessel water level. The procedure also described the addition of indicating lights at the main control panels to indicate loss of power to the valve lockup circuit.

2.7.2 Reactor Core Isolation Cooling System

On September 2, the initial reactor core isolation cooling system (RCIC) turbine overspeed trip was the result of inadequate procedural instruction for manual system initiation. The procedure directed the system be manually started with pump discharge flow directed through the minimum flow line. When RCIC pump discharge pressure equalled reactor vessel pressure, the procedure directed transfer to reactor vessel injection flowpath. The RCIC flow control system flow element is located downstream of the minimum flow line. Therefore, when the system was manually initiated the flow element sensed essentially no flow and demanded increased RCIC turbine speed to increase pump discharge flow until the turbine reached an overspeed condition and tripped. This event was the first manual initiation of the RCIC system since plant procedure 2.2.22, "Reactor Core Isolation Cooling System," was revised to reduce potential high/low pressure system interface events. The procedure had been validated on the plant specific simulator, however the simulator fidelity did not reflect flow element configuration. The procedure was revised to return to the original system operating mode for manual initiation and injection to the reactor vessel. Also the procedure was revised to discuss the proper operation and resetting of the RCIC turbine overspeed device and swap over from the vessel injection to full flow test modes and vise versa.

2.7.3 Startup Feedwater Regulating Valve

Plant procedure 2.4.49, "Loss of Normal Feed and Feedwater Control Malfunction," was revised to provide guidance to the operators upon the failure of the startup feedwater regulating valve in the open position with resultant increasing reactor vessel level. The use of reactor feed pump minimum flow valves to control reactor level was also modified.

2.7.4 High Pressure Coolant Injection

Plant procedure 2.2.21, "High Pressure Coolant Injection System," was revised to provide additional instruction to the operators concerning low system flow situations with flow control in automatic.

2.7.5 Shutdown Cooling System

Plant procedure 2.2.19, "Residual Heat Removal," is being revised to include instructions for operation of the keep-fill system during startup of the shutdown cooling system to reduce the potential for hydrodynamic events. Inspector review of the draft revision determined proposed changes were appropriate.

2.8 Reactor Restart from Forced Outage

On September 17, the licensee made preparations for reactor startup from the unscheduled outage. At 7:23 p.m. an automatic Group I PCIS actuation occurred immediately after the shutdown cooling system (SDC) was secured causing closure of the main steam isolation valves (MSIVs). The isolation occurred due to a high reactor vessel water level condition which resulted from; initially high reactor vessel water level prior to securing SDC; higher than normal initial reactor coolant system temperature; and the tripping of both residual heat removal (RHR) pumps and closure of the RHR injection valve in very close order.In response to the Group I isolation, the SDC system was returned to service and the event was reviewed to ensure all components responded as designed. Following satisfactory event review the isolation was reset, the SDC system was secured, and plant startup was resumed.

On September 18 at 3:42 a.m., reactor criticality was achieved and the reactor was stabilized for several days at approximately 120 psi and 1% of rated power to support post work and operability testing of the turbine driven HPCI and RCIC systems. Following successful completion of the testing, reactor pressure was increased to approximately 400 psi to support temporary leak seal repair of the MO-1001-50 SDC system suction valve. On September 24 repairs to MO-1001-50 were completed and ascension to full power was commenced. The turbine generator was synchronized to the distribution system on September 25 and the reactor achieved 100% of rated power on September 28.

The operations staff performed well during reactor restart. Response and corrective actions to the Group 1 isolation were appropriate. Plant conditions were well maintained throughout HPCI and RCIC system testing evolutions.

3.0 <u>Maintenance and Surveillance</u> (IP 37828, 61726, 62703, 93702)

3.1 <u>Malfunctions and Failures Associated with September 2 Scram</u>

The September 2 manual reactor scram was initiated as a result of a component failure in the non-safety related feedwater regulating valve (FRV) control air system. Additionally, several system malfunctions and component failures created operational challenges in maneuvering the reactor to a cold shutdown condition. Each malfunction and failure including failure mechanism, causal analysis, and licensee corrective actions enacted or planned, is addressed below individually.

3.1.1 <u>Loss of Feedwater Regulating Valve Position Control</u>

Loss of FRV position control which resulted in an increasing reactor vessel water level was the initiating cause of the event. The FRV control was lost due to a blown fuse (1/8 amp) in a common FRV control air power supply (640-42). The blown fuse caused both FRVs to lockup in fixed position. However, as control air decayed and bled from the FRV lockup system, the valves tended to slowly stroke open under the opening spring force.

Additionally, the blown fuse caused control room FRV lockup light indication to be inoperable. The fuse blew when moisture from the previous steam leak on the "B" FRV penetrated a junction box on the "A" FRV control air panel, migrated internal to conduit, and collected in a pressure switch (PS-656A) causing it to ground and short. The pressure switch was powered by the circuit (640-42) protected by the fuse.

Beyond the previous "B" FRV packing leak which introduced abnormal amounts of moisture into the surrounding area, the major contributing factor to the fuse failure included the degraded material condition of the junction box seal. Visual inspection of the junction box revealed evidence of previous standing water indicating the degraded junction box was a long standing condition.

As was previously noted, the feedwater control system and the FRVs are not safety related. As such, power supplies are not required to be independent and separate and junction boxes are not subject to environmental qualification criteria.

The licensee replaced the failed pressure switch. The degraded junction box was refurbished and weep holes were drilled in all FRV junction boxes to prevent water buildup. The junction box maintenance program practices were also reviewed.

Additionally, the licensee implemented a modification (PDC 90-56) to provide separate power supplies (from the 640-42 power supply) for each FRV, with each output having an internal protective fuse (1/8 amp). This modification provides the capability for continued operation of one FRV in the event a fuse should fail in the circuitry of the other FRV control circuit.

3.1.2 Loss of Control of the Startup Feedwater Regulating Valve

Following the reactor scram, operators attempted to utilize the startup feedwater regulating valve (non-safety related) to provide reactor vessel water level control. However, a degraded internal diaphragm caused the air booster relay to fail which resulted in a startup FRV lockup, similar in effect to the main FRV lockup but independent in cause. Feed pump discharge pressure forced the valve full open thereby delivering approximately 8-10% of full power feedwater flow to the reactor vessel.

Probable cause for the booster relay failure was believed to be random end of life component failure with contribution from a harsh operational environment. However, exact in service and shelf life times for the booster relay were not able to be determined. The failed booster relay was replaced. Additionally, the valve was repacked with live load packing to provide enhanced stem leakage mitigation potential. Following these activities the valve was successfully stroke tested.

3.1.3 Repetitive Reactor Core Isolation Cooling System Trips

Operators attempted to manually initiate the reactor core isolation cooling (RCIC) system three times following the manual reactor scram to provide reactor vessel water level control. However, the RCIC turbine tripped shortly after each attempted system initiation. The first turbine trip occurred as a result of an actual overspeed condition due to an inadequate procedure to control manual initiation and is discussed in Section 2.7.2.

The second and third RCIC system turbine trips where different from the first in that an actual overspeed condition was not achieved. On the second start attempt, injection valve MO-1301-49 was opened and for approximately twelve seconds flow to the reactor vessel was established before the trip and throttle valve actuated. The injection valve has a fifteen second stroke time. While this valve was closing, the injection check valve 1301-50 remained approximately 1 inch off the close seat causing the RCIC pump suction line to be pressurized until the injection valve stroked full closed (analysis of this condition is documented in sections 6.1 and 6.2).

On the third and final start attempt, RCIC operated for approximately 80 seconds in the full flow test mode until a spurious overspeed trip occurred prior to opening of the injection valve.

The premature overspeed tripping of the RCIC turbine on the second and third start sequences were the result of excessive mechanical tolerances in the trip and throttle valve linkage caused by past multiple actuations and normal system vibrations and were exasperated by previous inadequate maintenance. Additionally, the internal tappet (valve) latch surface was slightly rounded. Also, the tappet guide area was affected by the intrusion of foreign material into the oil system which affected reset capability.

The licensee enlisted vendor support to the resolution of the RCIC turbine trip events. The vendor representative immediately identified the excessive trip and throttle valve linkage tolerances, as well as, foreign material intrusion into the tappet guide and rounding of the tappet nut latch surface.

The trip and throttle valve was dissembled and cleaned as was the overspeed trip weight and reset spring assembly, and the tappet nut latch surface was machined to design specifications. The oil system was drained, the sumps were cleaned, and the system was refilled.

Following plant startup with the reactor stabilized at approximately 120 psi the licensee conducted extensive post maintenance RCIC testing. The RCIC turbine overspeed trip test verified proper function of the trip and throttle valve. Several delays were encountered while performing this test. Initially the test procedure necessitated revision to provide a discharge path from the RCIC barometric condenser to the suppression pool. Additionally, the EG-M (electronic governor module) and a damaged multi-point pin connector, which were contributing to erratic speed indication, were replaced.

Following completion of the RCIC turbine overspeed test, the revised RCIC system operating procedure (2.2.22) was performed and validated. Additionally, the licensee performed a Temporary Procedure (TP 90-68) which verified full flow test capability as well as verified full flow manual reactor vessel injection capability. This test also verified that RCIC discharge check valve 1301-50 fully seated. All RCIC testing was satisfactorily completed and the system was declared operable on September 24.

3.1.4 High Pressure Coolant Injection System Overspeed Trips and Erratic Automatic Operation

During recovery from the scram, operators manually initiated the high pressure coolant injection (HPCI) system two times. The first start was initiated to provide reactor vessel level control and was utilized for approximately two minutes. The second start was initiated to provide reactor vessel pressure control.

During both HPCI system starts the operators noted flow oscillations in automatic flow control at less than 3000 gpm. The oscillations were eliminated when the flow controller was placed in manual control.

Additionally, during multi-disciplinary analysis team review analysis of HPCI system data, the licensee determined on both initiations that the HPCI turbine experienced actual overspeed trips which automatically reset during the start sequences.

The licensee, with vendor support, conducted extensive causal analysis of each of the observed anomalies in HPCI operation, however was unable to ascertain definitive root cause determinations. With respect to the overspeed trips, the licensee identified a hand operated valve in the turbine oil system in the full open position that the vendor recommended be approximately one turn open. The valve (HO-2301-123), which is the oil relay pilot supply block valve that ports control oil to affect control valve responsiveness to position demand signals, was repositioned and appropriate procedures were revised to reflect the new position.

Licensee review of the HPCI flow oscillations while at low flow conditions (<3000 gpm) in automatic control identified that the EG-R actuator needle valve was one full turn open vice one-quarter turn open as recommended by the vendor. Adjustment of the needle valve provided smoother automatic control at low HPCI flow conditions. The licensee also attributed the oscillations to the design limitation of the automatic controller to maintain stable flow conditions at the low end of its operating range. Due to the presence of contaminants in the oil system, it was drained and flushed.

The licensee connected temporary diagnostic instrumentation to the HPCI system to accumulate additional operational data during post maintenance testing and subsequent surveillance testing. The post maintenance testing conducted September 23 indicated improved HPCI low flow operation as well as the absence of overspeed trip conditions and the system was declared operable. In order to expediently assess the long term effectiveness of the HPCI system corrective maintenance, the licensee has changed the HPCI surveillance periodicity from monthly to every two weeks for the next two months. However, due to the inconclusive root cause determinations for the HPCI system operational anomalies this issue is identified as an unresolved item until the results of testing have been reviewed (50-293/9^ ^0-01).

3.1.5 Shutdown Cooling System Suction Valve Failure to Open and Automatic System Isolation Actuation

On September 3 while performing initial shutdown cooling system (SDC) alignments, the inboard suction isolation valve, MO-1001-50 did not stroke full open in response to control room operator signal. The valve is designed as a seal-in valve and should stroke to full open position following a single control switch manipulation. The valve was subsequently opened by holding the control switch to the open position until full open position indication was received.

The licensee was unable to duplicate the seal-in circuit function failure during troubleshooting. However, fluctuating seal-in relay contact resistance values indicated potential contact degradation. The seal-in relay was replaced. Additionally, the valve's circuit breaker open coil and auxiliary contractor was replaced.

Additionally, while entering shutdown cooling on September 3, an automatic Group III PCIS actuation occurred. The isolation occurred due to a sensed reactor pressure of greater than 100 psig upon SDC initiation. The PCIS pressure sensing switch signal originates in the recirculation loop suction piping in close proximity to the SDC connection. It should be noted these isolations have been a recurring problem.

Licensee investigation, in conjunction with the vendor, indicated the potential for steam bubble formation in the SDC suction piping between the recirculation loop and MO-1001-50. Collapse of a steam bubble upon SDC initiation could provide a momentary pressure pulse sufficient to cause Group III actuations. This position is supported by generic industry (BWR) experience. Due to SDC/RHR piping configuration the potential exists for incomplete system vent and fill evolutions. Additionally, a pressure switch instrument line snubber was observed to be missing, which may have contributed to the isolation events.

The pressure switches were calibrated by the licensee and determined to be within acceptable limits. Additionally, SDC operating procedure (2.2.19) was revised to provide improved vent and fill instruction and to provide slower suction piping backfill via the keepfill system. The licensee is reviewing a 1976 vendor service letter addressing RHR/recirculation system water hammer during plant cooldown.

This issue was addressed previously in NRC inspection report 50-293/90-15. The inspectors will continue to observe and review licensee activity during SDC initiation.

3.1.6 High Safety Valve Tailpipe Temperature

Shortly before the September 2 reactor scram, operators noted that safety valve RV-203-4B tailpipe temperature recorders indicated an increase in temperature from 165 to 185 degrees F. The safety valve actuation setpoint is 1240± 12 psig. Reactor pressure was 1036 psig prior to the scram and decreased following the scram. Following containment de-inerting the licensee visually inspected the safety valve and tailpipe without observing evidence of actuation or leakage. On September 22, a drywell tour with the reactor at approximately 120 psi was conducted. A minor packing leak on hand operated root valve HO-1301-90 was observed to be impinging on the safety valve tailpipe causing the elevated temperatures. The leak was terminated by tightening the valve packing gland and tailpipe temperatures returned to normal.

3.2 Maintenance Program Improvements

As was documented in Special Inspection Report 50-293/90-21, insufficient maintenance supervision was present during the RCIC trip and throttle valve disassembly and the FRV packing removal to provide oversight and ensure as found data was properly recorded. Additionally, instances of inadequate procedural instruction and work area lighting were also observed by NRC inspectors during conduct of these activities. The report noted the recent licensee self-assessments of the maintenance program which identified several deficiencies that were evidenced during this event. Further, the report recommended the licensee accelerate programmatic improvements resultant from the self assessments.

During the September 12 management meeting the licensee discussed conceptual aspects of the maintenance improvement program. Subsequent to the meeting the licensee provided the inspector with a detailed summary of the major elements of the program.

Major elements of the licensee maintenance program improvement plan completed or scheduled include:

- Implementation of the Work Control Process as written to relieve the first line supervisor of administrative functions. To aid in this effort, two system engineers and three NED engineers were temporarily reassigned to the work control group.

- First line supervisors were directed to perform duties only as prescribed in their job description, with the major emphasis on job-site supervision.

- The production work force has been restructured into teams, with each team assigned to one specific first line supervisor. The supervisor will be responsible for team performance, training, qualification and personal accountability. Each team will be specifically assigned complete responsibility for routine repairs, preventive maintenance and surveillances.

- A "quality of workmanship" training module is under preparation by the Nuclear Training Department. The module is intended to provide specialized training in the completion of quality work activities and should be ready early in 1991.

- The "Rework Program" effort between systems and maintenance to identify rework activity, determine root cause and initiate corrective actions is planned to be implemented before November, 1990.

- Improved maintenance request scheduling on a quarterly basis.

- A "Maintenance Quality Improvement Program" is planned to be developed and implemented in which anyone who had involvement in selected jobs will participate in critiques chaired by the maintenance manager. Strengths and weaknesses will be determined. Lessons learned will be developed and incorporated into maintenance processes.

- The Preventive Maintenance Program is planned to be upgraded Four predictive maintenance programs are planned to be in place in early 1991; including thermography, lube oil analysis, vibration monitoring and analysis, and bearing temperature trending.

- The maintenance procedure upgrade program continues to make progress. The program includes the normal technical verification and human engineering as well as such items as a plant impact statement, identification repair and replacement parts for preventive maintenance and surveillances, proposed tagouts, radiological work permits, and special tools. This effort should complete in 1992.

In addition to routine inspector review of maintenance performance, the effectiveness of the improvement program will be comprehensively assessed during the upcoming NRC maintenance team inspection to be conducted November 5-16, 1990.

4.0 Emergency Preparedness

The inspector conducted a complete review of the criteria of emergency plan implementing procedure EP-IP-100, "Emergency Classification," Revision 1 with respect to the events of the September 2 manual reactor scram to independently determine if any emergency action level entry conditions were satisfied. The inspector concluded that the operational occurrences of the September 2 manual reactor scram did not present emergency action level entry conditions and therefore the licensee was appropriate in not activating the emergency plan.

Additionally, the inspector concluded the licensee appropriately implemented the instruction of procedure EP-AD-130, "Responsibilities of On Call Management Representatives," Revision 2. The procedure directs that the on call EP manager notify Commonwealth and town officials of events which are significant enough to warrant increased awareness of plant management but are not serious enough to warrant implementation of the emergency plan via emergency action level classification. Inspector review of procedure notification checklists indicated all specified state and local officials or their alternates were either contacted or attempted to be contacted during the morning of September 3.

5.0 Safety Assessment and Quality Verification

5.1 Multi-Disciplinary Analysis Team

Immediately during recovery from the September 2 manual reactor scram the licensee classified the event as a Type 3 scram consistent with the instruction of procedure 1.3.37, "Post Trip Reviews." A Type 3 scram is an event in which the course of the scram is not positively known and/or some safety related or important equipment functioned abnormally during the event. In the instance of the September 2 event the course of the scram was understood, however, the high pressure core injection and reactor core isolation cooling system malfunctions were not readily understood.

A Type 3 reactor scram classification also requires that a multi-disciplinary analysis team (MDAT) be assembled to review each aspect of the event and identify root cause determinations.

The licensee established a 38 member MDAT on September 3. The MDAT team leader was the station chief technical engineer who possesses a current senior reactor operator license. The 37 team members were engineers and technical specialists that represented diverse plant disciplines. The MDAT was stationed in the technical support center and provided oversight during day and evening shifts. Extensive assessment of MDAT performance is documented in the Special Inspection Report (50-293/90-21). The MDAT continued to function effectively through plant restart. The MDAT issued a comprehensive report that was submitted to the Operations Review Committee (ORC) for review and approval prior to restart. The ORC provided extensive review and analysis of the MDAT report and subsequently recommended plant startup readiness to the Station Director. All MDAT report followup items were serialized and were required to be reviewed and approved by ORC prior to closure.

Subsequent to plant restart the MDAT amended applicable report sections to incorporate the results of HPCI post maintenance testing. However, amended MDAT report sub- sections were not presented to ORC in a timely fashion. This inspector concern was brought to the attention of the MDAT team leader who acted to ensure ORC was provided report updates in a timely manner.

5.2 LER 90-10

LER 90-10, "Completion of a Shutdown Due to One Inoperable Recirculation Loop," addressed the July 3, 1990, TS required plant shutdown necessitated by the inability to restart the "A" M-G set which had tripped the preceding day. This event was documented in NRC inspection report 50-293/90-15. The LER accurately detailed the event; provided extensive technical description of the recirculation system and described corrective actions enacted.

The ORC review of this LER was noteworthy. The failure and malfunction report (F&MR 90-204) which will document the event root cause determination was not complete when the LER was drafted. Therefore, the initial LER presented to ORC lacked sufficient causal analysis. The ORC quorum appropriately noted this weakness and requested further causal analysis development be included in the LER. The LER was revised to include documentation of a modification to the M-G set speed control circuitry (PDC 90-14) and was presented to ORC. Subsequently, ORC approved the LER with the addition of a commitment to submit a supplement after the F&MR investigation is completed. The licensee expected the supplement to be submitted by November 2, 1990. The inspector had no further concerns regarding this LER.

5.3 LER 90-11

LER 90-11, "Automatic Closing of the PCIS Group III Isolation Valves While Shutdown," addressed the July 3, 1990 automatic isolation of the shutdown cooling (SDC) system immediately following system initiation. This event was documented in the NRC Inspection Report 50-293/90-15. The LER appropriately addressed the reporting criteria as well as identified similar previous events. An additional similar event is documented in Section 5.6 of this report.

As discussed in the referenced inspection report, the inspector will review the status of the plant design change intended to enhance SDC venting. The inspector had no questions regarding this LER.

5.4 LER 90-12

LER 90-12, "Two Radioactive Sources Not Leak Checked Within TS Required Interval," addressed the July 12, 1990 licensee discovery that two of twenty-six radioactive sources had not been leak checked within the six month periodicity required by TS.

Upon discovery, the two sources, a 200 microcurie (Uci) cesium-137 source and a 5.8 Uci americium-241 source, were immediately inventoried and leak checked satisfactorily (less than minimal detectable activity). The missed source check was identified while the licensee was preparing for a Quality Assurance audit. The event was the result of a procedural weakness. The radioactive test source procedure (6.6-010) did not provide a consolidated list of sources maintained by the radiological operations support (ROSD) and chemistry divisions (CD). The procedure was performed by the ROSD and the missed sources were used by the CD. The procedure is being revised to ensure effective control of radioactive sources. The LER fully addressed the reporting criteria including a previous similar event. The inspector had no further questions regarding this LER.

5.5 LER 90-13

LER 90-13, "Manual Scram Due to Lockup of the Feedwater Regulating Valves," addressed the September 2, 1990 feedwater regulating valve control system failure which necessitated a manual scram from approximately 100% power and the subsequent equipment and functions that occurred when the licensee attempted to reach the cold shutdown condition. This event has been documented in the inspection report as well as Special Inspection Report 50-293/90-21. The LER accurately detailed the event and provided accurate technical descriptions of the individual events. The inspector had no further questions regarding this LER.

5.6 LER 90-14

LER 90-14, "Automatic Closing of the Pri[illegible]ntainment System Group III Isolation Valves While Shutdown," address[illegible] September 3, 1990 automatic isolation of the shutdown cooling (SDC) system immediately following system initiation. This event is documented in section 3.1.5 of this inspection report. The LER appropriately addressed the reporting criteria as well as identified similar previous events. The inspector had no questions regarding this LER.

5.7 Review of Periodic and Special Reports

Upon receipt, the inspector reviewed periodic and special reports submitted pursuant to Technical Specifications. This review verified, as applicable: (1) that the reported information was valid and included the NRC-required data; (2) that test results and supporting information; and (3) that planned corrective actions were adequate for resolution of the problem. The inspector also ascertained whether any reported information should be classified as an abnormal occurrence. The following reports were reviewed:

-- Monthly Operational Status Summaries for August and September 1990.
-- Operations Review Committee and Nuclear Safety Review and Audit Committee Meeting Minutes.

6.0 Engineering and Technical Support

6.1 Reactor Core Isolation Cooling Discharge Check Valve Design Deficiency

The reactor core isolation cooling discharge check valve CK-1301-50 failed to fully close following the second RCIC trip during the September 2 event causing a momentary pressurization of the RCIC pump suction piping. Engineering analysis of the affected piping is documented in Section 6.2.

Licensee as found field observations indicated the check valve was approximately 1 inch from the close seat. Minimal torque was applied to the external test shaft to fully seat the valve. The licensee, with vendor support, disassembled the valve and conducted a detailed component identification and dimensional specification verification.

Check valve CK-1301-50 is a four inch carbon steel testable swing check valve manufactured by the Anchor/Darling Valve Company. Licensee root cause investigation identified a valve design deficiency as the failure mechanism which prevented full valve closure. The valve shaft key which provides torque translation capability for external testing was observed to be loose in its keyway and also was observed to be contacting the valve packing and bushing. This interference caused increased frictional force contributing to the inability of the valve to fully close. Additionally, a knob or dog which could be utilized to provide valve position indication (but is not utilized) was also determined to have the potential to impede valve closure.

The licensee modified the shaft key and securely staked it to the shaft. The key length was reduced decreasing key to packing and bushing interference potential. The knob on the shaft bushing was also removed.

The vendor concurred with the licensee determination of a valve design deficiency as well as with the licensee modifications to the valve. The licensee is currently assessing the reportability of this design deficiency with respect to the criteria of 10 CFR 21. The licensee indicated that as a minimum a letter is planned to be submitted to the NRC describing the observed deficiency.

During the previous April 12, 1989 RCIC pump suction piping pressurization event, the check valve also failed to properly seat. The cause of that failure was determined to have been shaft binding caused by the lodging of residual leak seal material between the valve body and shaft bushing. NRC Inspection Report 50-293/89-80 provided detailed documentation of the April 12, 1989 event. Therefore, although the failure of the CK-1301-50 valve to properly seat occurred in both events, causal analysis determined the failure mechanisms of each event were not closely related.

The licensee demonstrated sound root cause investigation processes in the development of the design deficiency determination. As-found data was properly observed and recorded. The corrective modifications were effectively implemented and received vendor concurrence. Startup RCIC system testing included full flow injection to the reactor vessel which successfully demonstrated that the check valve properly opened and fully seated. The inspector had no further questions.

6.2 Analysis of Reactor Core Isolation Cooling System Suction Piping Pressurization

As was previously discussed, the reactor core isolation cooling (RCIC) system suction piping experienced a momentary pressurization condition when the RCIC discharge check valve (1301-50) failed to fully seat after the second unsuccessful system start attempt following the September 2 manual reactor scram. The overpressure condition was not noted during the event but rather was identified several days later during MDAT evaluation of the emergency plant information computer (EPIC) charts of the September 2 RCIC start sequences. The RCIC pump suction pressure trace indicated approximately a 90 psi per second pressure increase immediately following the RCIC turbine trip. The pressure transient was less than fifteen seconds in duration and was terminated by the closure of RCIC injection valve MO-1301-49. Additionally, EPIC data indicated that the suction thermal relief valve PS-1301-31 lifted and relieved to the floor drain system as designed.

The low pressure RCIC system suction piping is six inch diameter, schedule 40 A-106B carbon steel with a minimum allowable stress of 15,000 psi, a minimum required yield strength of 35,000 psi and an ultimate tensile strength of 60,000 psi. The suction piping design service conditions are 80 psig and 170 degrees F. The RCIC system was designed to ASME B31.1 code criteria.

On April 12, 1989, the licensee experienced a similar but more severe RCIC system suction piping pressurization event. During the April 1989 event, the suction piping was instantaneously overpressurized which resulted in a "sonic wave affect" or hydrodynamic pulse within the piping system. The engineering analysis of the April 1989 event assumed a 0.001 second pressurization time with a peak pressure of 900 psi. The current pressurization event pressure rise rate was 90.38 psi per second based on a calculation of the slope of the EPIC RCIC pump suction pressure trace. At this rate it would take a pressurization time of approximately ten seconds to reach the postulated 900 psi suction piping pressure. However, the licensee conservatively assumed full pressurization occurred in one-half second (0.5 seconds) which would still result in a pressurization time that would be 500 times slower than that the of April 1989 event. Therefore, review of the current event data and related calculations clearly indicated a RCIC system suction piping "sonic wave affect" from instantaneous overpressurization did not occur. Additionally, licensee calculations determined that the circumferential and longitudinal pressures combined with deadweight stresses were well within design basis code allowable stress.

In conjunction with the above calculations, the licensee conducted a visual inspection of the structural integrity of the RCIC system which revealed no damage. Magnetic particle examination was performed at the calculated highest stress weld area on the suction piping which identified only manufacturing defect indications.

Based on the above calculations, analysis, and examinations the inspector concluded that RCIC system suction piping design specifications were not exceeded by the September 2 pressure transient.

6.3 Leak Repair of Shutdown Cooling Suction Isolation Valve MO-1001-50

During the course of the September 2-17, unscheduled outage, the licensee applied a temporary leak repair technique to the shutdown cooling system inboard suction isolation valve, MO-1001-50. Previously, the valve was experiencing a pressure seal ring area leak. On September 4, 1990 the licensee issued maintenance request (MR 90-10-73) to initiate the repair activities. The temporary repair was to be accomplished by the injection of a leak sealant compound in accordance with the directions of Plant Design Change (PDC) 89-49 and its revisions, consistent with this associated safety evaluation.

Two key parameters of the safety evaluation included the initial reactor coolant system (RCS) pressure necessary to facilitate sealant compound injection and the maximum volume of leak sealant compound to be injected. The initial RCS pressure was significant to prevent sealant compound from entering the RCS. The safety evaluation required a minimum RCS pressure of 200 psig prior to injection. The maximum volume of sealant compound to be injected was significant to ensure it would not exceed the volume of the voided pressure seal ring area and potentially infiltrate the RCS.

Nuclear engineering department procedure, 3.0.3, "Field Revision Notices," requires cognizant division managers or designated engineers to review field revision notices to PDCs to determine if such changes are major or minor and safety related or non-safety related. The procedure defines a major FRN as one that: changes the original conceptual design or the intent of the implementation document, or affects the bases on which the approved safety evaluation was made.

Contrary to the procedure above, the licensee issued and dispositioned, as minor, two FRNs to PDC 89-49 that affected the bases of the original safety evaluation. On September 11, the licensee issued minor field revision notice FRN 89-49-15, which among other things, authorized the injection of sealant compound under cold conditions of 25 +/- 5 psig. On September 23, the licensee issued minor FRN 89-49-19, which authorized injection of an additional 100 sticks of leak sealant which caused the total authorized volume of sealant compound to be injected to exceed the voided pressure seal ring volume.

Upon licensee identification of these issues, Failure and Malfunction Report (F&MR) 90-340 was issued on September 23, 1990. The engineering department revised the safety evaluation to PDC 89-49 to analyze the impact of the two FRNs of concern. The evaluation utilized the conservative assumption that all sealant compound injected entered the valve bonnet void area. The total volume of sealant injected was 197 cubic inches and the voided bonnet area volume was approximately 8,000 cubic inches. The evaluation concluded that the relative ratios of the two volumes, in conjunction with close tolerance clearances the sealant compound would have to traverse to enter the bonnet void area, are such that the operation of the valve was not impacted.

The engineering department initiated an aggressive investigation into this event utilizing a department critique matrix. The initial investigation results determined that the major nature of the two FRNs of concern was not identified due to personnel error on the part of the reviewing cognizant engineer. Engineering management developed a lesson plan to reinforce FRN review criteria and a department quality memo was issued. A deputy division manager was also assigned to the fluid system and mechanical components division to improve supervisory oversight. Additionally, the NED division manager instituted a policy to review all temporary leak seal PDCs and FRNs.

Notwithstanding licensee response to this event, the failure to properly evaluate and approve this design change is considered a violation (50-293/90-20-02).

7.0 Management Meetings (IP 30702, 30703)

7.1 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 October 31, 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.

7.2 "A" Recirculation Pump Motor-Generator Set Conference Call

On August 23, a conference call was conducted between technical staff members of NRR, NRC Region I and the licensee to summarize licensee actions in response to the repetitive automatic trips of the "A" recirculation pump motor generator set experienced since October 1989, and to update current status. An NRC synopsis of this issue was provided in Inspection Report 50-293/90-15.

7.3 Conference Calls and Meetings in Response to the September 2-3 Event

On September 4, NRC Region I management initiated a conference call with the licensee to express concern with component failures and system malfunctions that presented operational challenges prior to and following the September 2 manual reactor scram. Region I was provided a current status of the licensee multi-disciplinary analysis team investigation. Additionally, NRC management informed the licensee that a Region I special inspection team was being dispatched to the site to review the event.

On September 7, following completion of the special inspection, NRC Region I management initiated a conference call with the licensee to discuss scheduling of a Management Meeting to address the event and to gain an agreement from the licensee to maintain the plant in a shutdown status until the meeting was conducted. The licensee provided agreement and the management meeting was scheduled for September 12.

On September 12, NRC Meeting Number M-90-114 was convened in the Region I office to discuss BECo corrective actions and restart planning related to the September 2-3 event. A list of attendees is included as Attachment I to this report. Mr. Charles Hehl, NRC Region I Director of Reactor Projects, opened the meeting with a brief introduction of the topic followed by a detailed description of the NRC concerns relative to the event. Mr. Ralph G. Bird, BECo Senior Vice President of Nuclear Operations, provided an overview of the licensee planned presentation. BECo representatives distributed the prepared overhead displays which are included as Attachment II to this report.

The licensee then made a presentation of the September 2-3 manual reactor scram and plant shutdown. The presentation included a detailed review of initial plant conditions, immediate operator actions and extended operator actions to maneuver the plant to cold shutdown. The licensee described the MDAT composition, charter, and activities and conclusions to that point in time. Additionally, the licensee provided a comprehensive review of each component failure and system malfunction; identified failure mechanisms and causal determinations which included programmatic implications; and identified short and long term corrective actions.

The licensee concluded the MDAT investigation continued to be functioning effectively. Additionally, senior licensee management stated that MDAT function and plant startup were dependent on satisfactory task completion not by critical date scheduling. The licensee projected plant readiness for restart during the upcoming weekend (September 15-16). A period of NRC questioning and requests for clarification and explanation followed the licensee presentation.

At the conclusion of the meeting, Mr. Hehl requested the licensee extend the agreement to not restart the plant until a conference call could be conducted on September 14 at which time the licensee would present a summation of the MDAT investigation and an Operations Review Committee recommendation to the Station Director to restart. The licensee provided agreement and the meeting was adjourned.

On September 14, NRC Region I management initiated the scheduled conference call with the licensee. The licensee presented closure of the remaining key MDAT items and identified six minor items to be completed prior to startup. The NRC Region I management concurred with licensee action and terminated the agreement to maintain the plant in shutdown. The licensee subsequently completed the six open items and presented them to the resident inspector before plant startup on September 17.

ATTACHMENT I
September 12, 1990
Meeting Attendees
Boston Edison Company - NRC

BECo

R. Anderson, Plant Manager
R. Bird, Senior Vice President - Nuclear
W. Clancy, Acting Technical Section Manager
G. Davis, Vice President - Nuclear Administrator
R. Fairbank, Nuclear Engineering Department Manager
L. Olivier, Operations Section Manager
E. Robinson, Manager Nuclear Information Division
G. Stubbs, Maintenance Section Manager
R. Swanson, Regulatory Affairs Manager

NRC/NRR

R. Conte, Chief BWR Section, Division of Reactor Safety (DRS)
C. Hehl, Director, Division of Reactor Projects (DRP)
W. Hodges, Director, DRS
J. Macdonald, SRI Pilgrim
E. McCabe, Acting Chief, Projects Branch 3, DRP
T. Martin, Regional Administrator
J. Rogge, Chief, Projects Section 3A, DRP
R. Wessman, Project Directorate I-III, NRR

Boston Edison Company

Pilgrim Nuclear Power Station

September 2, 1990 - Plant Shutdown

ATTACHMENT 11

Discussion Topics

- Findings of Multi-Disciplinary Analysis Team (MDAT)
- Basis for Confidence in Plant Equipment
- Conclusion and Projections for Restart

Plant And Personnel Safety Was Maintained Throughout The Event

- Water Level Transient Minor and Well Controlled
- Operators Stabilized the Plant and Responded Properly to Equipment Malfunctions
- Operators Performed Well

MDAT Focus Is Appropriate

- Identified Key Problems
- Root Causes
- Not Constrained by Schedule
- Access to Available Resources
- ORC, NMC Oversight

Feed Regulating Valve Control System Identified As Source Of Transients

- Feedwater Regulating Valve "B" Packing
- **Feedwater Reg Valve Lockup**
- Startup Feedwater Regulating Valve Air Booster Relay

RCIC ISSUES

- Overspeed Trip Mechanism
- Trip Throttle Valve Linkage
- Injection Check Valve

HPCI ISSUES

- Flow Controller
- Overspeed Trip

RHR/SHUTDOWN COOLING ISSUES

- Manual Control Circuit - Inboard Shutdown Cooling Suction Valve

- Automatic Shutdown Cooling Isolation

ACTIONS THAT HAVE BEEN/WILL BE ACCOMPLISHED PRIOR TO RESTART

- RCIC Oil Flush/Inspect
- RCIC Linkage Disassembly/Adjustment
- HPCI Oil Sump Clean/Flush
- RCIC Check Valve Repair/Test
- Shutdown Cooling Valve Relay/Contact Replacement
- Correct FRV Water Damage

ACTIONS THAT HAVE BEEN/WILL BE ACCOMPLISHED PRIOR TO RESTART (cont'd)

- **FRV Repacked**
- **RCIC Suction Pipe Inspection**
- **Procedure Revisions**
- **Appropriate Restart Testing**

We Have Confidence In System Reliability

- Workmanship Performance Review
- Successful Loss of Offsite Power Test
- Maintenance Personnel Qualification Review
- **System Engineer Review**
- Augmented Testing

Acccelerate Ongoing Long Term Improvements

- Predictive Maintenance
- Preventive Maintenance
- Technical Support

CONCLUSION

- MDAT Properly Focused and Thorough
- Equipment Issues Being Resolved
- Confidence in System Reliability
- Accelerate On-Going Program Improvements

ENCLOSURE 2

STAFF ANALYSIS OF EVENT CLASSIFICATION

ITEM 1 - Emergency Core Cooling System (ECCS) Initiation

In response to this concern, it should be noted that none of the conditions for initiating a notification of an Unusual Event contained in the licensee's symptom-based Emergency Action Level (EAL) procedures were met during the September 2, 1990 event. Therefore the NRC concluded that the licensee was in compliance with their procedures in not initiating a notification of an Unusual Event. The staff also has determined that the licensee's EALs meet the intent of the guidance of NUREG-0654. In addition, it is the NRC's position that the manual initiation of HPCI, as occurred during the September 2, 1990 event, when none of the automatic initiation set-points were in jeopardy of being exceeded, does not meet the intent of NUREG-0654 for initiation of a notification of an Unusual Event.

ITEM 2 - Indications or Alarms

This is a non-safety related indication of a condition on a non-safety related system. The function of the lockup lights is to indicate to the control room operators that due to an apparent malfunction in the control air system, the FRVs have been locked in place in their current position. The manual reactor scram was initiated on September 2, not because control room indication of an FRV lockup condition was lost, but rather because of a failure in the feedwater control system which created a high reactor vessel water level condition that the operators were unable to correct.

ITEM 3 - Other Plant Conditions

In stating your concern, you cite licensee entry into various emergency operating procedures (EOPs) during the course of the September 2 event, and you note that additional licensee personnel responded to the plant during the event. This EAL guideline provides very general guidance as well as discretion to licensees. During and following reactor scrams, it is not unusual for licensees to enter and exit the EOPs as necessary, nor is it unusual for licensees to enlist additional resources to assist in plant recovery from scram events.

During the September 2 event, the licensee determined that the operational occurrences of the manual reactor scram did not present EAL entry conditions, and therefore the emergency plan was not activated. However, the licensee did implement an administrative procedure which directs that Commonwealth and town officials be notified of events which are significant enough to warrant increased plant management awareness but are not serious enough to warrant emergency plan activation via emergency action level classification.