U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.: 50-293

Report No.: 50-293/88-27

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

Facility: Pilgrim Nuclear Power Station

Location: Plymouth, Massachusetts

Dates: July 18, 1988 - August 28, 1988

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Areas Inspected: Routine resident inspection of licensee action on previous inspection findings, plant operations, radiation protection, physical security, plant events, maintenance, surveillance, and outage activities. In addition, the status of the licensee's power ascension test program was reviewed. Principal licensee management representatives contacted are listed in Attachment I to this report.

Results:

Unresolved Item: The licensee identified an incorrectly installed valve yoke clamp on the "B" core spray full flow test return line isolation valve. The licensee's corrective actions, including inspection of the yoke clamps on similar safety-related valves, will be reviewed in a future inspection (Section 3.b, UNR 88-27-01).

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DETAILS

1.0 Summary of Facility Activities

The plant has been shut down for maintenance and to make program improvements since April 12, 1986. The reactor core was completely defueled on February 13, 1987 to facilitate extensive maintenance and modification of plant equipment. The licensee completed fuel reload on October 14, 1987. Reinstallation of the reactor vessel internal components and the vessel head was followed by completion of the reactor vessel hydrostatic test. The primary containment integrated leak rate test was also completed during the week of December 21, 1987.

During this report period, the NRC conducted the Integrated Assessment Team Inspection (IATI) on August 8 through August 24, 1988 to determine the readiness of licensee management, staff, and programs to support power operation at the facility. The fifteen member team was composed of both NRC Region I and the Office of Nuclear Reactor Regulation (NRR) personnel, as well as two observers from the Commonwealth of Massachusetts. The results of the IATI are documented in Inspection Report 50-293/88-21.

On August 24, 1988, Mr. William T. Russell, Region I Administrator, was onsite and attended the exit interview for the IATI. A Systematic Assessment of Licensee Performance (SALP) management meeting for the Pilgrim facility was held on August 25, 1988, in Plymouth, Massachusetts to discuss the results of SALP Board Report number 50-293/87-99. On August 26, 1988, the NRC Advisory Committee on Reactor Safeguards conducted a publicly held subcommittee meeting on Pilgrim at the Plymouth Memorial Hall in Plymouth, Massachusetts.

2.0 Followup on Previous Inspection Findings

Violations

(Closed) Violation (87-16-01), Failure to Perform Fire Protection System Surveillances. During inspection 50-293/87-16 the inspector identified four instances of failure to properly implement surveillances of fire protection equipment. In three cases the individuals performing the surveillance indicated that equipment conditions were acceptable when in fact outstanding maintenance requests documented unacceptable conditions. Calculations performed during the fourth surveillance were improperly completed.

In response to the violation the licensee's Operations Section Manager issued a department memo describing the incidents and providing additional guidance on the conduct of fire protection surveillances. The contents of this memorandum were discussed onshift with operations personnel by licensee management. The licensee also revised the format and content of the surveillance procedures to eliminate confusion regarding their intent.

The inspector reviewed the revised procedures and the operations department memorandum, and discussed these items with a sample of operators. Several recently completed surveillances were reviewed. The inspector found the results to be accurate and acceptable. During this review the inspector noted that Surveillance Procedure 2.1.12, "Daily Diesel Generator Surveillance," Step 62, states that to verify emergency diesel generator room fire suppression system heat trace operability, check the local temperature indicator or the local power available light. The inspector questioned the acceptability of checking only the light in that it is not a true indication of heat trace operability. In response the licensee initiated a procedure change to require verification of both the temperature indicator and the light. The inspector also noted that during a check of heat trace operability performed in response to the violation, discrepancies with circuit drawing E697 were noted. The inspector questioned the current adequacy of the drawing. The licensee subsequently initiated an Engineering Service Request to verify the drawing accuracy. Since issuance of the violation, the resident inspectors have routinely evaluated fire protection equipment status and surveillance testing with no additional problems noted. The inspector had no further questions.

(Closed) Violation (87-45-02), Failure to Properly Implement QC Receipt Requirements. Unresolved item 87-34-02 was opened when it was noted that the licensee had installed three nonconforming drywell spray nozzles (out of 220). The licensee purchased the nozzles as non "Q" and upgraded them to "Q" status using their Commercial Quality Item (CQI) procurement process. Further inspection revealed that only 32 of the 220 drywell nozzles had undergone receipt inspection, while the licensee's Quality Assurance Manual requires 100% receipt inspection. This incident was caused by the lack of clear guidance for performing receipt inspections, and by differences between licensee and contractor (Bechtel) QC programs. In reviewing the licensee's corrective actions, the inspector noted that while corrective actions were pursued, the root cause was not established by the licensee until prompted by the inspector. Violation 87-45-02 was subsequently issued because the licensee had not identified the 100% receipt inspection requirement specified for all CQI that are to be upgraded to "Q" status.

Subsequent to the three nonconforming nozzles being replaced, the licensee issued a memorandum to their contractor (Bechtel Construction, Inc.) that more clearly specified receipt inspection requirements for commercial quality items. On a more permanent basis, Bechtel was requested to revise their Quality Control Instruction Manual to ensure that 100% receipt inspection is performed for CQI items. In addition several Bechtel administrative procedures were changed to improve instructions regarding receipt inspection requirements for commercial quality items. The inspector reviewed these procedures and found them to be adequate and to unambiguously require 100% receipt inspection of a CQI. Licensee procedures changed as a result of this incident include Revision 8 to CQI 7.01, Receipt Inspection, which also requires 100% receipt inspection of CQIs unless otherwise specified in writing. With these procedures in effect, it appears the licensee has taken adequate corrective action.

The licensee undertook an investigation to establish the proper disposition all CQI purchases received on site since inception of the CQI program in 1984. It was found that there were 487 CQI purchases that required review to determine proper disposition. The receipt inspection form for each of these 487 was reviewed to ensure that an appropriate sampling plan and acceptance criteria had been specified. A written technical justification was prepared for each CQI to document acceptable results of this review. The inspector found that the licensee has taken adequate corrective actions to preclude recurrence. Further, the licensee has reviewed and verified the acceptable resolution of all CQIs since the CQI program began in 1984.

Unresolved Items

(Closed) Unresolved Item (85-30-10), Lack of Adequate Design Analysis for Replacement of Valve 2301-74. A Safety System Functional Inspection (SSFI) conducted at Pilgrim (50-293/85-30) identified that the design analysis performed when the HPCI turbine exhaust stop check valve was replaced, was inadequate. The new valve had a disc lift pressure about three times higher than that of the original valve. The higher lift pressure appeared to reduce the ability of the installed vacuum breaker to function. No 10CFR50.59 safety review was conducted as a part of the design process, because this modification was characterized as an upgrade.

Subsequent to the SSFI a plant modification (PDC 85-59) was designed and installed. This modification added piping and an increased capacity vacuum breaker between the HPCI turbine exhaust pipe and the torus. The connection of the new vacuum breaker line is downstream of check valve 2301-74 at a high point in the HPCI exhaust piping just before it enters the torus. Because of the new location of the vacuum breaker the increased lift pressure of valve 2301-74 will have no impact on the proper operation of the HPCI exhaust line vacuum breaker. Additionally, the licensee has performed an analysis which shows that although the increased lift pressure of the new valve (2301-74) does increase the HPCI turbine backpressure, it is within the manufacturers limits. The inspector had no further questions.

(Closed) Unresolved Item (85-30-11), Failure to Perform Adequate Design Analysis in Support of Removal of Insulation from the RHR Heat Exchanger. Plant Design Change 84-75 was reviewed by the NRC Safety System Functional Inspection (SSFI) team. The purpose of this modification was to remove insulation from the Residual Heat Removal (RHR) system heat exchanger to facilitate inspection. The review identified a number of deficiencies in the calculations which could affect the results and thereby render the conclusions concerning the removal of the insulation questionable. A concern was raised that the calculation did not adequately demonstrate that the capacity of the safety-related heating, ventilating and air conditioning equipment in the RHR equipment rooms was sufficient to maintain design temperatures.

Calculation M-518-1, Revision 0, August 19, 1986 was prepared in response to this SSFI concern. The new calculation, which was reviewed by the inspector, addresses the NRC SSFI comments as follows:

- The new calculation now references the original calculation for RHR compartment cooling, and all significant heat loads including piping and electrical were considered.

- An average shell temperature for shutdown cooling from the manufacturers heat exchanger data sheet was used.

- The cooling coil specification/data sheet identifies the 115 degrees F design temperature and is referenced.

- Actual motor efficiencies were used.

The calculation shows that removal of the insulation will result in a temperature increase of less than 1 degree F in the area, and for normal operation the temperature will be within the 115 degrees F design. The inspector had no further questions.

(Closed) Unresolved Item (86-34-02), Evaluate the Quality Designation of the Refueling Bridge Interlocks. During inspection 50-293/86-34 the inspector questioned the licensee's decision to classify the reactor refueling bridge refueling interlocks as a non-quality (Q) item. The inspector collected information concerning the refueling interlocks design basis, and forwarded the information to the NRC Office of Nuclear Reactor Regulation (NRR) for technical review. NRR subsequently completed this technical review and concluded that classification of the interlocks as non-Q was acceptable. This conclusion is documented in a NRR evaluation dated June 25, 1987. The inspector also noted that the licensee has designated the refueling bridge as a "Management Q" item as defined by the Boston Edison Quality Assurance Manual. This designation requires that quality verification measures be applied to design changes, maintenance and surveillance affecting the components.

(Closed) Unresolved Item (87-45-01), Control of Drawings. During this inspection the adequacy of corrective actions implemented in response to deficiencies identified in the Document Control program at Pilgrim were reviewed. The inspector verified that licensee corrective actions to Quality Assurance Deficiency Report (DR) 1700 have been effectively implemented by the licensee, conducted an interview with a Pilgrim Document Control Clerk (PDCC), and reviewed the current instructions on the use of the aperture card file in the Document Control Center. It was noted that the PDCC had been trained to disregard annotated messages on aperature cards and to use the computer (BOS-SEEK) data base for current messages. In addition, apprpriate caution notes were prominently posted on the Aperature Card File. The inspector also noted that Records Management Group Work Instruction Number 2.32, "Verification of PDC/FRN Annotations", contains paragraphs that require plant design change packages (PDC) and field revision notices (FRNs) be checked to verify outstanding messages in the SEEK database. Although the work instructions specify a minimum sample size, the inspector learned that as a practice, all drawings in every PDC/FRN are checked for correct messages. To reinforce this, Work Instruction Number 1.04 requires documented training of all records management division personnel in the performance of their assigned tasks. The inspector noted that both the DCC supervisor and DCC clerk have received this training.

The inspector reviewed all open PDC packages for 1988 and found evidence of complete review by Document Control. In addition, the inspector performed a re-inspection of controlled drawing stick files in the plant. Thirty-six drawings were checked, with a total of 93 messages. The drawings and messages were compared against a current CD/CDA drawing run-off. All messages and drawing revision numbers were in agreement between the computer printout and each drawing in the stick file. The inspector found that the licensee has taken adequate corrective action to preclude recurrence of the deficiencies noted in DR 1700 and unresolved item 87-45-01.

(Update) Unresolved Item (87-45-04), Use of Appendix G to the FSAR to Support Degrees of System Operability. Inspection Report 50-293/87-45, Section 4 described the licensee's intended use of Appendix G of the Final Safety Analysis Report (FSAR) to declare the Standby Gas Treatment System (SBGTS) operable for fuel load operations only (conditionally operable). This interpretation took into account plant conditions at that time and concluded that the SBGTS was capable of performing its design functions for the existing plant conditions. The inspectors questioned the licensee on the practice of using Appendix G in this manner. The licensee subsequently agreed to make SBGTS fully operable for all modes of plant operations prior to the start of refueling, and suspend use of Appendix G in this application pending further NRC review.

Discussions between the inspector and the licensee's Nuclear Engineering Department Manager have confirmed the licensee's commitment not to use Appendix G in the determination of conditional system operability when making plant changes to a more restrictive mode. This item will remain open pending additional inspector review of the use and applicability of FSAR Appendix G.

(Update) Unresolved Item (88-07-02), Inadequate Post Work Testing of E-203 Work. Inspection Report 50-293/88-07, Section 4.f, describes the licensee's ongoing effort to replace or repair numerous deficiencies in plant electrical systems. The project is now essentially complete and the licensee has conducted satisfactory retests on 46 percent of the circuitry involved. The inspector will review the licensee's decision regarding the need to perform additional testing during a future inspection.

(Closed) Unresolved Item (88-07-03), Inadequate Station Overtime Control. Inspection Report 50-293/88-07, Section 5.0, detailed weaknesses in the licensee's program to control overtime use by station personnel. The inspector reviewed the licensee's recently issued procedure for control of overtime and found that the procedure meets the recommendations of Generic Letters 82-12 and 83-14. The inspector identified a weakness in the procedure in that it exempted personnel who work on site for less than thirty days. The licensee was made aware of this weakness and has committed to apply the procedural requirements to all persons involved in safety-related work regardless of the length of stay on site.

Implementation of overtime controls has been observed in all functional areas. The use of overtime is now being controlled in a uniform manner throughout the organization. Satisfactory implementation was also confirmed during the Integrated Assessment Team Inspection. Based on reviews of licensee records, documented correspondence and interviews with licensee personnel, the inspector concluded that the current overtime control program at Pilgrim station is satisfactory.

(Update) Unresolved Item (88-11-02), Human Factors - Related Problems with Emergency Operating Procedures (EOP) and Their Satellite Procedures. During inspection 50-293/88-11 problems with EOP satellite procedures were identified such as a lack of clarity, consistency and attention to detail. Typical examples were differences between labeling as presented in the procedure and as actually appearing on control room panels, unclear directions presented to operators and the location, availability and control of tools or equipment specified in procedures. Although the procedures had been issued prior to the inspector's review, the licensee stated that the final walkthrough of each procedure was not yet completed. Accordingly, this was identified as an unresolved item pending completion of the licensee's validation effort.

During the current period the inspector selected Procedure 5.3.21, "Bypassing Selected Interlocks", Revision 6, as a representative example of a procedure to be used with EOP. It was noted that panel and terminal strips were adequately marked and labeled. The inspector also noted that although emergency lighting might not be sufficient to accomplish jumpering of connections deep in the back of several panels, procedure step B.1 identifies the location of a tool box (including flashlights) expressly provided for this purpose. The tool box was inspected and found to be in good order and well controlled by the Watch Engineer.

A second procedure selected for review was Procedure 2.2.25, "Fire Water Supply System", Revisison 9. The inspector noted that the procedure identifies valve MOV 3479 as "High Pressure Feedwater Heater Train A Downstream Block Valve", while the valve switchplate reads "Train A HP HTr Downstream Block Valve." Further inspection found an inconsistency in the way the satellite procedures were reviewed and validated/verified. Attachment A to Procedure 1.3.4-15, Revision 0, is the common validation/verification checklist used when reviewing satellite procedures. Step 17 states, "Are equipment numbers and nomenclature used in the procedure identical to those which are displayed on the equipment?" A review of the completed checklist for this procedure found the reviewer's initials in the "Yes" column. The licensee investigated this issue and found a difference of opinion as to the intent of "identical" in the validation/verification checklist. While licensee management intended identical to mean verbatim, the operators/technical persons completing the checklist interpreted "Identical" to mean close enough so that there would be little chance of mis-operation.

As a result of this finding, the licensee undertook a review of all satellite procedures to assure that "equipment numbers and nomenclature" were identical between procedures and switchplates. Twenty-eight procedures were found to require minor typographical or editorial changes. These procedures are being corrected and will be reissued during August, 1988. The balance of the satellite procedures having differences between switchplate and procedure description (41) do not warrant change because the valves in question are major, front panel, remote-manual valves most often identified by operators using their valve number only. Since valve number is the key identifier and is used most often in describing these valves, any changes could introduce unwanted uncertainty in operator actions. This item will remain open pending completion of licensee actions and additional review by the inspector.

Inspector Follow Items

(Closed) Inspector Followup Item (84-03-05), Review the Licensee's CRD Rebuild Work Records. During 1984 the inspector questioned the adequacy of licensee control rod drive mechanism (CRDM) rebuild records. Checklists used for two CRDMs did not include all required signatures. An extensive CRDM rebuild program was conducted during refueling outage (RFO) 6. The inspector noted that subsequently the required CRDM scram timing and friction testing were successfully conducted. The licensee maintained the unit in operation for about 15 months without significant CRDM problems. Based on this operating experience and successful testing, the adequacy of rebuild activities during RFO 6 appear adequate. The resident inspectors have observed CRDM overhauls during the current refueling outage and noted that they were effectively controlled.

<u>(Closed) Inspector Followup Item (85-20-15). Inconsistencies in the MOV Thermal Overload Heater Selection Criteria.</u> A Safety System Functional Inspection (SSFI) conducted at Pilgrim Station (50-293/85-30) identified that the licensee had not established a criteria for selection of overload heater elements for motor operated valves (MOV). This resulted in overload heaters being selected with a range of protection from 78% to 193% of motor full load current for identical motors. Although overload heaters for MOV's are only used for alarm purposes, this condition could result in motor damage going undetected. Subsequent to the SSFI a licensee study was conducted which resulted in calculations (PS-38) which established overload heater selection criteria, and the required size for each motor. Approximately ten motor overload heaters have been changed, to comply with the sizes specified in PS-38. The inspector reviewed calculation PS-38 in detail and it appears to be consistant with established MOV manufacturer and IEEE criteria. The inspector had no further questions.

<u>(Closed) Inspector Followup Item (86-06-06), Review the Cause and Corrective Actions Associated with RCIC Residual Flow Indication.</u> During the conduct of the reactor core isolating cooling (RCIC) pump operability tests conducted in February and March of 1986 an anomolous condition was observed by the NRC inspectors. An intermittent condition resulted in a residual flow indication of approximately 50 GPM, on the flow indicating controller, after the operability test. The flow transmitter (FT 1360-4) that feeds the flow indicating controller (FIC 1340-1) was functionally tested, and the residual flow indication was duplicated during a RCIC flow test. Root cause analysis conducted by the licensee, has determined that the cause of the anomalous indication was due to draining of the flow transmitter reference legs during removal and installation of the flow orifice. Pilgrim Nuclear Power Station Procedure 8.5.5.3, "RCIC Flow Rate Test at $\leq$ 150 PSIG," has been revised to require I&C to backfill the flow transmitter after completion of the test. The backfill method proved successful on 3 subsequent runs of the RCIC system. The inspector had no further questions.

<u>(Closed) Inspector Followup Item (86-38-04), Licensee to Take Action to Assure That at Least One Individual on Each Shift Could Operate the Fire Truck.</u> The inspector noted that the licensee had no formal, routine training program on the station fire truck, and that the on-shift fire brigade leader had not operated the truck pumping unit in several years and could not operate it during the demonstration. The licensee has developed a training module on the fire truck, including a practical demonstration. Additionally, training has been provided to operations personnel on the use of the fire truck; the licensee intends to provide this training on an annual basis. The inspector had no further questions.

3.0 Routine Periodic Inspections

The inspectors routinely toured the facility during normal and backshift hours to assess general plant and equipment conditions, housekeeping, and adherence to fire protection, security and radiological control measures. Inspections were conducted on weekends on July 24, 31, and on August 27, 1988 for 8 hours. In addition, substantial backshift and weekend inspections were conducted during the period as part of the Integrated Assessment Team Inspection. Ongoing work activities were monitored to verify that they were being conducted in accordance with approved administrative and technical procedures, and that proper communications with the control room staff had been established. The inspector observed valve, instrument and electrical equipment lineups in the field to ensure that they were consistent with system operability requirements and operating procedures.

During tours of the control room the inspectors verified proper staffing, access control and operator attentiveness. Adherence to procedures and limiting conditions for operations were evaluated. The inspectors examined equipment lineup and operability, instrument traces and status of control room annunciators. Various control room logs and other available licensee documentation were reviewed.

The inspector observed and reviewed outage, maintenance and problem investigation activities to verify compliance with regulations, procedures, codes and standards. Involvement of QA/QC, safety tag use, personnel qualifications, fire protection precautions, retest requirements, and reportability were assessed.

The inspector observed surveillance and post-work tests to verify performance in accordance with approved procedures and LCO's, collection of valid test results, removal and restoration of equipment, and deficiency review and resolution.

Radiological controls were observed on a routine basis during the reporting period. Standard industry radiological work practices, conformance to radiological control procedures and 10 CFR Part 20 requirements were observed. Independent surveys of radiological boundaries and random surveys of nonradiological points throughout the facility were taken by the inspector.

Checks were made to determine whether security conditions met regulatory requirements, the physical security plan, and approved procedures. Those checks included security staffing, protected and vital area barriers, perronnel identification, access control, badging, and compensatory measures when required.

a. System Alignment Inspection

On August 3, 1988, the inspector walked down portions of "A" and "B" residual heat removal system to confirm that system lineup procedures match plant drawings and the as-built configuration. This walkdown was also conducted to identify equipment conditions that might degrade performance, to determine that instrumentation is calibrated and functioning, and to verify that breakers and valves are properly positioned and locked as appropriate. No discrepancies were identified by the inspector.

b. Plant Maintenance and Outage Activities

Rotation of the Yoke on Core Spray MO-1400-4B Valve Body

On August 16, 1988, licensee identified during a system walkdown that the yoke of the "B" core spray valve full flow test return line isolation valve (MO-1400-4B) had rotated out of the correct orientation. Valve MO-1400-4B is a motor operated gate valve. The yoke is held to the valve body by a yoke clamp which was found to be installed incorrectly. The licensee's investigation determined that an inadequate procedure and maintenance personnel error during valve maintenance in August, 1987 had contributed to the discrepancy.

The inspector reviewed the maintenance request package and the quality control (QC) inspection report associated with the valve maintenance in August, 1987. It appears that the yoke clamp was installed incorrectly due to the symmetrical appearance of the yoke clamp and lack of match marking during disassembly. The licensee took immediate action to correctly reinstall the yoke clamp. At the close of this inspection, a separate maintenance request (MR), MR 88-14-48, was initiated to inspect "A" core spray test return line isolation valve MO-1400-4A. An inspection of similar safety-related valves is in progress to verify the proper orientation of the yoke clamp. The licensee also initiated a revision to maintenance procedure 3.M.4-10, "Valve Maintenance", which will require match marking and labeling valve components during disassembly and verifying the same upon reassembly. This item will remain unresolved pending further review (88-27-01).

Update on Repair of Two Residual Heat Removal System Valve Yokes

On June 7, 1988, the licensee discovered cracking in a motor operated valve (MOV) yoke in the Residual Heat Removal System (RHR). The RHR system consists of two redundant loops with two pumps per loop. During operation of the Low Pressure Cooland Injection (LPCI) mode of the RHR system each loop injects to the reactor vessel through a single line at the reactor recirculation system. In addition to serving as LPCI injection paths the two lines serve as flow paths for shutdown cooling return. Each injection line contains a motor operated globe valve, a motor operated gate valve and a check valve in series. While attempting to remove the "B" loop of shutdown cooling from service the licensee was unable to secure flow when the globe valve, MOV 1001-28B, was closed. Followup inspections by the operations staff identified that the yoke had cracked about 270 degrees around at a weld between the lower yoke section and the motor actuator mounting plate. Subsequent inspection of the counterpart valve in the "A" RHR loop, MOV 1001-28A, identified indications of cracking in the lower portion of the yoke, just below the location of the crack in the MOV 1001-28B yoke. The motor operators for the two valves are Limitorque SMB-5 actuators mounted to uniquely modified yokes. No other SMB-5 actuators or similar yoke designs exist at Pilgrim.

The material analysis conducted by the Massachusetts Institute of Technology (MIT) on the damaged valve yoke from MOV-1001-28B revealed that the failure point on the yoke was at a weld located on the upper portion of the yoke near where it is mounted to the valve operator. Analysis of the weld revealed a lack of fusion of the weld to the base metal of the yoke. In addition, analysis of the force exerted on the yoke by the valve operator revealed that the yoke experienced a significantly higher stress value during operation than had been originally expected by the designers. The combination of these two factors led to the failure of the valve yoke at the point noted.

The licensee has since modified the design of the yokes to significantly increase their strength. These changes include shortening the neck of the yoke and redesigning the yoke weld which had previously failed, thereby reducing the stress level experienced at that point. The licensee also inspected and reassembled the valve operators and adjusted the torque switch settings on the MOV to reduce the maximum stress experienced by the valve during cycling. At the close of the inspection, the licensee was in the process of conducting post-maintenance testing on the valves and MOVATS testing on the valve operators. This item remains unresolved (UNR 88-25-01) pending review of the test results, the MIT failure analysis report, the design changes to the yoke and licensee evaluation of the failure contributors for applicability to other valves.

c. Physical Security

On August 16, 1988, the onsite security contractor for the Pilgrim Nuclear Power Station was changed from the Globe Security Systems to the Wackenhut Corporation. Licensee planning for the transition and the effectiveness of licensee management's control during the transition was reviewed by a safeguards specialist inspector during the Integrated Assessment Team Inspection. It was determined that the change in the contract security force was accomplished without any compromise of security and with minimal disruption to security operations. Details of the inspector's assessment are described in inspection report 50-293/88-21.

4.0 Review of Plant Events

The inspectors followed up on events occurring during the period to determine if licensee response was thorough and effective. Independent reviews of the events were conducted to verify the accuracy and completeness of licensee information.

a. Deficient Solenoid Valves Installed in Safety Systems

On July 19, 1988, licensee engineering personnel concluded that solenoid valves installed in four applications at the Pilgrim station were not adequate to perform their intnnded function. NRC Information Notice (IN) 88-24, "Failures of Air Operated Valves Affecting Safety-Related Systems," describes potential problems with various types of air operated valves. During their review of IN 88-24, the licensee identified four applications in which the maximum differential operating pressure of the solenoid valves was was not adequate for the application. The station instrument air system normally operates at about 110 psig. The air supply pressure to the solenoid valves in question is controlled by a non-safety related pressure regulator. Failure of the pressure regulator would subject the solenoid valves to a differential pressure of 110 psid. The maximum differential pressure rating of the solenoid valves is 40 psid. The excessive pressure differential could prohibit the valve from operating.

Two of the deficient solenoid valves are installed in the Control Room High Efficiency Air Filtration System and one is installed in the Standby Gas Treatment System for damper control. The remaining solenoid valve provides motive air to the inboard containm . isolation valve on a reactor water sample line. The licensee informed the NRC via ENS of the problem. A plant design change (PDC) has been initiated to replace the valves with qualified components. This PDC is scheduled for implementation prior to restart.

b. A Small Fire Inside the Protected Area

On August 2, 1988, a small fire occurred inside the protected area. Repairs to the permanent sodium hypochlorite storage tank located in the intake structure were ongoing. As part of these repairs, several bolts were being removed using a torch. At 3:25 p.m., the assigned fire watch observed smoke escaping from beneath fire retardant cloth which had been installed to support the repair work. The room was evacuated, the onsite fire brigade assembled, and the fire was extinguished a brief time later. In accordance with the licensee procedures, the Plymouth Fire Department was notified; however, their assistance was not required. Two fire trucks arrived onsite at about 3:50 p.m., after the fire had been declared out. The fire occurred in a building outside the radiologically controlled area of the plant. Consequently, there were no radiation or radioactive contamination hazards involved. The licensee notified the NRC via ENS of the incident at 5:30 p.m. on August 2, 1988.

5.0 Power Ascension Test Program (PATP)

The inspector reviewed the current status of the licensee's Power Ascension Test Program. In addition, specific issues identified during the Management Meeting on April 8, 1988 were reviewed and are discussed below.

- Power Ascension Program Startup Test Review

  The inspector reviewed the licensee's Draft Initial Report regarding their Power Ascension Program Startup Test Review during inspection 50-293/88-14. The licensee's final report "Independent Review of the Adequacy of Power Ascension Program Testing," dated June 1988 was reviewed and discussed with licensee representatives during this inspection. The inspector found the licensee's analysis of the affect of plant modifications on the dynamic response of the plant to be acceptable. However, the inspector was concerned regarding the deficiencies identified during this review in the post- modification testing for several plant design changes (PDC) including changes to SBGTS and ADS Logic (PDC's 86-70 and 86-73). The licensee stated that two PCAQs (Potential Conditions Affecting Quality) have been issued to Engineering and Modification Management specifically addressing the reasons for the discrepancies in testing. Resolution of the PCAQs will be reviewed in a future resident inspection.

- Rosemount Transmitters

  The inspector reviewed BECo Letter #88-117, dated August 4, 1988, regarding Rosemount transmitter "ringing". This letter was in response to NRC questions regarding potential "ringing" associated with Rosemount transmitters and the need to instrument the level transmitters during the PATP to adequately detect "ringing" problems. The licensee states that the Nuclear Engineering Department performed an evaluation of the Rosemount transmitters installed at Pilgrim and has concluded that Pilgrim's applications are not susceptible to a significant "ringing" problem. In regards to instrumenting the transmitters during the PATP, the licensee states that monitoring will be performed during the PATP if the EPIC system is operational, otherwise it will be performed during power operations following the PATP. Following discussion with licensee representatives regarding their response, it was determined that further NRC:RI review of the justification would be required to adequately resolve this issue. Subsequent to the end of the inspection period, this issue was resolved in a Management Meeting on August 31, 1988, as documented in a letter from Mr. T. T. Martin (NRC:Region I) to Mr. R. G. Bird (BECo), dated September 7, 1988.

- Overall PATP

  The inspector discussed the status of the overall PATP with the Assistant Startup Test Manager. The inspector reviewed the latest approved revisions of the licensee's PATP procedures and verified that previous NRC comments were incorporated. In addition the inspector reviewed Station Instruction SI-SG.1025, "Independent Review of Test Results," dated June 15, 1988 which details the independent review process to be used during the PATP. The inspector questioned the Assistant Startup Manager regarding the specific responsibilities of the Systems Engineering Division Manager and the Startup Test Manager during the independent review process. The Assistant Startup Manager stated that he would revise the Station Instruction to further detail their responsibilities. He also indicated that he is currently in the process of assigning system engineers to perform the individual review function for each power ascension test.

- Findings

  No unacceptable conditions were identified. Items requiring further followup, identified during the Management Meeting of April 8, 1988, have been resolved.

6.0 Management Meetings

At periodic intervals during the course of the inspection period, meetings were held with senior facility management to discuss the inspection scope and preliminary findings of the inspectors. A final inspection exit interview was conducted on September 6, 1988. No written material was given to the licensee that was not previously available to the public.

A final exit meeting for the Integrated Assessment Team Inspection (IATI) 50-293/88-21 was held onsite on August 24, 1988 to discuss the findings with the licensee senior management.

On August 25, 1988, a Systematic Assessment of Licensee Performance (SALP) management meeting was held in Plymouth, Massachusetts. The meeting was open to the public and results of the SALP 50-293/87-99 covering the period of February 1, 1987 to May 15, 1988 were discussed.

NRC staff members from Region I and NRR attended a licensee briefing on August 17, 1988, to discuss the results of risk assessment studies conducted at Pilgrim Station. A handout was presented by the licensee and is attached to this report (Attachment 2).

Attachment I to Inspection Report 50-293/88-27

Persons Contacted

R. Bird, Senior Vice President - Nuclear
* K. Highfill, Station Director
R. Anderson, Plant Manager
E. Kraft, Plant Support Department Manager
A. Morisi, Acting Planning and Outage Department Manager
D. Swanson, Nuclear Engineering Department Manager
J. Alexander, Plant Operations Section Manager
J. Jens, Radiological Section Manager
J. Seery, Technical Section Manager
R. Sherry, Maintenance Section Manager
P. Mastrangelo, Chief Operating Engineer
D. Long, Security Section Manager
W. Clancy, Systems Engineering Division Manager
F. Wozniak, Fire Protection Division Manager

*Senior licensee representative present at the exit meeting.

COMPARISON OF PILGRIM-SPECIFIC PSA & IPE WITH RESPECT TO CONTAINMENT VENTING

Two quantitative evaluations of the effects of containment venting have been performed using Pilgrim-specific modeling.

The Pilgrim interim PSA results were first presented at the Containment Performance Workshop in February, 1988. In March, 1988 the results from a preliminary version of the Pilgrim IPE were presented during an NRC inspection. This particular IPE Model has been specifically developed for the purpose of evaluating containment venting in response to requests made by the Staff in their August 21, 1987 letter to Boston Edison on this subject.

A summary of the results from these two studies follows:

	Frequency of Venting
PSA	1.5E-4/Yr.
IPE	2.9E-4/Yr.

As noted above, these evaluations yield similar results with respect to the expected frequency of containment venting. The difference between the two studies is on the order of a factor of two, which is good agreement between two probabilistic based studies performed using relatively independent methods.

The differences between the two evaluations is a result of independent modeling assumptions made in each study with respect to the availability of systems and equipment to remove decay heat from containment and provide core cooling during events in which containment heat removal systems are assumed to be lost. Some of the more significant assumptions follow.

The following differences tend to increase the likelihood of venting, or core damage as modeled in the IPE.

- o The availability of RHR is less in the IPE than the PSA as a result of assumptions associated with instrument and control effects on the suppression pool cooling mode of RHR (LPCI interlocks) and more conservative common cause modeling.
- o The reliability of the containment vent system in the IPE is less than the PSA due to human error modeling which incorporated factors associated with reluctance to vent on the part of the plant staff or from authorities outside the plant which are not included in the PSA models.

The following differences tend to decrease the likelihood of venting, or core damage as modeled in the IPE.

- o Recovery of systems important to containment heat removal are explicitly included in the IPE.
- o The main condenser is assumed not to be available for a wide variety of PSA transients including loss of feedwater and MSIV closure events.

- Use of drywell sprays with the fire system is considered as a means of containment pressure control in the Pilgrim IPE.

- The PSA conservatively assumes that on containment failure due to overpressure, inadequate core cooling occurs. The IPE credits systems external to the reactor building which would not be subject to the potentially harsh environment within the reactor building following containment failure.

The results of both studies indicate that the conditions which may lead to venting are infrequent (on the order of once in several thousand years of operation). They both also indicate that initiating the containment vent under the explicit conditions specified in the Pilgrim EOPs has relatively significant beneficial effect on overall core damage probability by minimizing containment heat removal as a contributor to risk at Pilgrim.

Even considering the modeling differences between the Pilgrim IPE and PSA, these two evaluations come to the same conclusions with respect to the likelihood of containment venting and its benefits with respect to preventing core damage.

PRELIMINARY

Table 1

PILGRIM DIRECT TORUS VENT SENSITIVITY STUDY
—SUMMARY OF RESULTS—

Frequency of Preventive Venting

TW Sequences
2.9E-4/Yr

Frequency of Mitigative Venting

Sequence Class	Containment PressureCtl	H_2 Ctl
TQUX	5.9E-9/Yr	9.1E-8/Yr
SBO	1.2E-6	2.3E-8
TQUV	2.8E-7	1.1E-8
AV	2.5E-6	1E-9
	1.5E-6/Yr	1.3E-7/Yr

Sequence	Core Damage Probability: Vent	Core Damage Probability: No Vent	Containment Release Probability: Vent	Containment Release Probability: No Vent	Dose Consequences*: Vent	Dose Consequences*: No Vent
TQUX	9.1E-6/Yr	9.1E-6/Yr	1.4E-7/Yr	1.3E-7/Yr	1.3 R/Yr	1.3 R/Yr
SPO	2.3E-5	2.3E-6	1.2E-6	5.5E-7	0.5	5.5
TQUV	1.1E-6	1.1E-6	2.8E-7	1.8E-7	0.1	1.8
TWQUV	2.1E-6	2.1E-5	2.1E-6	2.1E-5	63	630
AV	1E-7	1E-7	2.4E-8	1.6E-8	.01	.16
Other	1E-5	1E-5	1E-5	1E-5	200	200
	2.5E-5/Yr	4.4E-5/Yr	1.4E-5/Yr	3.2E-5/Yr	260 R/Yr	840 R/Yr

*Assumptions as to dose consequences by accident class (reference IDCOR 16.1 for Peach Bottom).

TW	3E+7R
Vent	4E+5R
ATWS	2E+7R
All others	1E+7R

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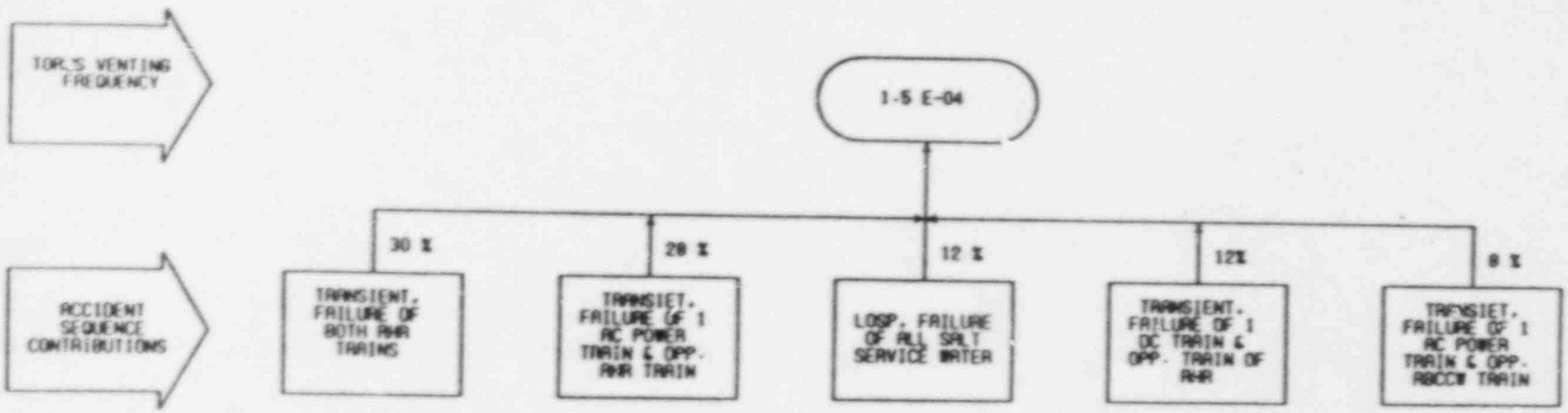
PILGRIM STATION PSA TORUS VENTING SEQUENCES



PILGRIM PLANT

CONTAINMENT HEAT REMOVAL SYSTEM RELIABILITY

DIFFERENCES BETWEEN IPE & PSA

IPE	PSA
VENT UNAVAILABILITY = 0.1/DEMAND	VENT UNAVAILABILITY = .01/DEMAND
RHR UNAVAILABILITY = 5.5E-4/DEMAND	RHR UNAVAILABILITY = 1.3E-5/DEMAND
RECOVERY OF FAILED SYSTEMS EXPLICITLY MODELED RHR MAIN CONDENSER MSIV CLOSURE SERVICE WATER	NO RECOVERY
MAIN CONDENSER AVAILABLE AS HEAT SINK IN ALL BUT LOSS OF MAIN CONDENSER LOSS OF OFFSITE POWER LOSS OF INSTRUMENT AIR STUCK OPEN SAFETY VALVE	NO MAIN CONDENSER ASSUMED FOR MSIV CLOSURE LOSS OF MAIN CONDENSER LOSS OF FEEDWATER LOSS OF OFFSITE POWER LOSS OF DC STUCK OPEN SAFETY VALVE
FIRE SYSTEM ALIGNMENT TO RHR ASSUMED TO BE CAPABLE OF CONTAINMENT PRESSURE CONTROL	FIRE SYSTEM NOT USED FOR CONTAINMENT CONTROL
SYSTEMS OUTSIDE REACTOR BUILDING CAPABLE OF MAKEUP TO REACTOR (CONDENSATE, FIRE SYSTEM) FOLLOWING CONTAINMENT FAILURE	CORE DAMAGE ASSUMED ON CONTAINMENT FAILURE

Table 1

PILGRIM DIRECT TORUS VENT SENSITIVITY STUDY *
— SUMMARY OF RESULTS —

Frequency of Preventive Venting

TW Sequences

~~2.9E-4~~/Yr
1.5E-4

Frequency of Mitigative Venting

Sequence Class	Containment Pressure Ctl	H_2 Ctl
TQUX	5.9E-9/Yr	9.1E-8/Yr
SBO	1.2E-6	2.3E-8
TQUV	2.8E-7	1.1E-8
AV	2.5E-8	1E-9
	1.5E-6/Yr	1.3E-7/Yr

	Core Damage Probability		Containment Release Probability		Dose Consequences*	
Sequence	Vent	No Vent	Vent	No Vent	Vent	No Vent
TQUX	9.1E-6/Yr	9.1E-6/Yr	1.4E-7/Yr	1.3E-7/Yr	1.3 R/Yr	1.3 R/Yr
SBO	2.3E-5	2.3E-6	1.2E-6	5.5E-7	0.5	5.5
TQUV	1.1E-6	1.1E-6	2.8E-7	1.8E-7	0.1	1.8
TWQUV	~~2.1E-6~~ 1.0E-6	~~2.1E-5~~ 1.0E-5	~~2.1E-6~~ 1.0E-6	~~2.1E-5~~ 1.0E-5	~~60~~ 30	~~600~~ 300
AV	1E-7	1E-7	2.4E-8	1.6E-8	.01	.16
Other	1E-5	1E-5	1E-5	1E-5	200	200
	~~2.5E-5~~/Yr 2.4E-5	~~4.4E-5~~/Yr 3.3E-5	~~1.4E-5~~/Yr 1.3E-5	~~3.2E-5~~/Yr 2.1E-5	~~260~~ R/Yr 230	~~840~~ R/Yr 510

*Assumptions as to dose consequences by accident class (reference IDCOR 16.1 for Peach Bottom).

TW	3E+7R
Vent	4E+5R
ATWS	2E+7R
All others	1E+7R

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* Modified to reflect PSA derived frequency of venting

PILGRIM PLANT
CONCLUSIONS OF PSA & IPE CONCERNING
EFFECTIVENESS OF DIRECT TORUS VENT

- PSA RESULTS

"RELIABLE TORUS VENTING IS IMPORTANT TO CORE DAMAGE PREVENTION"

REFERENCE: PRESENTATION TO NRC BWR MARK I CONTAINMENT WORKSHOP BY PICKARD, LOWE & GARRICK, INC.
FEBRUARY 25, 1988

- IPE RESULTS

"CONTAINMENT VENTING AS SPECIFIED BY EMERGENCY PROCEDURES HAS A LARGE BENEFICIAL EFFECT ON PLANT RISK BY EFFECTIVELY ELIMINATING CONTAINMENT HEAT REMOVAL FAILURE AS AN ACCIDENT CLASS"

REFERENCE: PRESENTATION TO NRC ON CONTAINMENT VENTING BY BOSTON EDISON
MARCH 7, 1988

CONCLUSIONS

IN SPITE OF THE MODELING DIFFERENCES USED TO QUANTIFY CORE DAMAGE FREQUENCY, THE IPE AND PSA COME TO THE SAME OVERALL CONCLUCIONS WITH RESPECT TO THE BENEFITS OF TORUS VENTING