U.S. NUCLEAR REGULATORY COMMISSION
REGION I

DCS Numbers
50-333-850322
50-333-850418
50-333-850421
50-333-850503
50-333-850220
50-333-850506

Report No. 85-09

Docket No. 50-333

License No. DPR-59 Priority -- Category C

Licensee: Power Authority of the State of New York
P.O. Box 41
Lycoming, New York 13093

Facility Name: J.A. FitzPatrick Nuclear Power Plant

Inspection At: Scriba, New York

Inspection Conducted: April 1, - May 31, 1985

Inspectors:
L.T. Doerflein, Senior Resident Inspector

W.J. Lazarus, Senior Emergency Preparedness Specialist

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Approved by: James C. Linville Jr. 6/25/85
James C. Linville, Chief, Reactor Projects Section 2C date

Inspection Summary:

Inspection on April 1, - May 31, 1985
(Report No. 50-333/85-09)

Areas Inspected: Routine and reactive inspection during day and backshift hours by two resident inspectors and one region based inspector (200 hours) of licensee action on previous inspection findings, licensee event report review, operational safety verification, surveillance observations, maintenance observations, plant startup from refueling, determination of reactor shutdown margin, startup testing of the Analog Transmitter Trip System, followup on

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licensee response to GE Service Information Letter No. 402, review of the Emergency Core Cooling Systems subject to potential overpressurization, follow-up on licensee event, relocation of the Emergency Operations Facility and review of periodic and special reports.

Results: No violations were identified in the areas inspected.

However, as discussed in paragraph 10, we are concerned about the failure to implement a modification on the Containment Atmosphere Dilution System to protect the carbon steel nitrogen makeup lines from low temperature brittle fracture. The significance of this modification was highlighted by the failure of the vent header at another facility, during the past year, due to improper operation of the nitrogen inerting system. We are also concerned that this maybe indicative of a general lack of progress in reducing the modification backlog identified in inspection report 50-333/82-24.

The continuing problems with pilot valve seat leakage and setpoint drift of the target Rock safety relief valves (discussed in paragraph 3) renew concerns regarding the need for increased management attention in pursuing resolution of these problems.

Other concerns involving Source Range Monitor and Intermediate Range Monitor instrument dry tube cracking, the Shutdown Margin demonstration, and the inadvertent lifting of a fuel bundle from the reactor core are documented in paragraphs 6., 8., and 12. respectively.

DETAILS

1. Persons Contacted

*R. Baker, Technical Services Superintendent
V. Childs, Senior Licensing Engineer
*R. Converse, Superintendent of Power
M. Curling, Training Superintendent
*W. Fernandez, Operations Superintendent
*H. Glovier, Resident Manager
H. Keith, Instrument and Control Superintendent
D. Lindsey, Assistant Operations Superintendent
R. Liseno, Maintenance Superintendent
*E. Mulcahey, Radiological & Environmental Services Superintendent
R. Patch, Quality Assurance Superintendent
T. Teifke, Security & Safety Superintendent

The inspector also interviewed other licensee personnel during this inspection including shift supervisors, administrative, operations, health physics, security, instrument and control, maintenance and contractor personnel.

*Denotes those present at the exit interview.

2. Licensee Action on Previous Inspection Findings

(Open) Unresolved Item (333/77-26-06): In a letter dated November 14, 1977, the architect-engineer indicated that the Containment Atmosphere Dilution System logic would be modified to provide low temperature protection for the carbon steel nitrogen makeup lines. The inspector noted that this modification has not been implemented. Additional details on this item are discussed in paragraph 10. of this report.

(Open) Inspector Followup Item (333/83-04-03): The inspector noted that the licensee continues to have problems with setpoint drift on the two stage Target Rock safety relief valves. Additional details on this item are discussed in paragraph 3. of this report.

3. Licensee Event Report (LER) Review

The inspector reviewed LER's to verify that the details of the events were clearly reported. The inspector determined that reporting requirements had been met, the report was adequate to assess the event, the cause appeared accurate and was supported by details, corrective actions appeared appropriate to correct the cause, the form was complete and generic applicability to other plants was not in question.

LER's 85-09*, 85-10*, 85-11, 85-12, 85-13* were reviewed.

*LER's selected for onsite followup.

LER's 85-09 and 85-13 reported that, when tested, a total of five Target Rock two stage safety relief valves had setpoints outside the Technical Specification allowable tolerance. The vendor believes that the possible causes of this setpoint drift are inadequate clearances in the laberinth seal area and pilot valve seat leakage. The vendor is paying particular attention to laberinth seal clearance during valve overhaul. The licensee was also informed by the vendor that the pilot seat leakage could be caused by testing the valves at to low a steam pressure such that the pilot valve doesn't have any cushion effect when shutting. As a result, the licensee revised the surveillance procedure to increase the test pressure to 250-300 psig. However, despite this change, following safety relief valve testing during the startup from the 1985 refueling outage, the licensee noted indications of pilot seat leakage on the "F" safety relief valve. The inspector will continue to review licensee's progress in resolving the safety relief valve drift during a subsequent inspection.

LER 85-10 reported that a fuel bundle was inadvertently lifted from the reactor core when it was caught on one of the lock levers of the fuel support grapple. Details of this event are discussed in paragraph 12. of this report.

4. Operational Safety Verification

a. Control Room Observations

Daily, the inspectors verified selected plant parameters and equipment availability to ensure compliance with limiting conditions for operation of the plant Technical Specifications. Selected lit annunciators were discussed with control room operators to verify that the reasons for them were understood and corrective action, if required, was being taken. The inspectors observed shift turnovers biweekly to ensure proper control room and shift manning. The inspectors directly observed the operations listed below to ensure adherence to approved procedures:

-- Reactor startup on May 28, 1985.

-- Issuance of RWP's and Work Request/Event/Deficiency forms.

No violations were identified.

b. Shift Logs and Operating Records

Selected shift logs and operating records were reviewed to obtain information on plant problems and operations, detect changes and trends in performance, detect possible conflicts with Technical

Specifications or regulatory requirements, determine that records are being maintained and reviewed as required, and assess the effectiveness of the communications provided by the logs.

No violations were identified.

c .Plant Tours

During the inspection period, the inspectors made observations and conducted tours of the plant. During the plant tours, the inspectors conducted a visual inspection of selected piping between containment and the isolation valves for leakage or leakage paths. This included verification that manual valves were shut, capped and locked when required and that motor operated valves were not mechanically blocked. The inspectors also checked fire protection, housekeeping/cleanliness, radiation protection, and physical security conditions to ensure compliance with plant procedures and regulatory requirements.

No violations were identified.

d. Tagout Verification

The inspector verified that the following safety-related protective tagout records (PTR's) were proper by observing the positions of breakers, switches and/or valves.

-- PTR 850548 on "C" Residual Heat Removal Service Water System.

-- PTR 850572 on "B" Station Battery Charger.

-- PTR 850603 on the Reactor Protection System.

-- PTR-850647 on the "A" Emergency Service Water System.

-- PTR 850783 on the "B" Residual Heat Removal System.

-- PTR 850858 on the "B" Residual Heat Removal Service Water System.

No violations were identified.

5. Surveillance Observations

The inspector observed portions of the surveillance procedures listed below to verify that the test instrumentation was properly calibrated, approved procedures were used, the work was performed by qualified personnel, limiting conditions for operation were met, and the system was correctly restored following the testing:

-- F-ST-39B, Type "B" and "C" LLRT of Containment Penetrations, Revision 14, dated March 20, 1985, performed April 8, 9 and 15, 1985.

-- F-ST-29E, Backup Scram Valves Functional Test, Revision 0, dated February 27, 1985, performed April 19, 1985.

-- F-ST-39L, Reactor Vessel Hydrostatic Test, Revision 0, dated May 1, 1985, performed May 7, 1985.

-- F-ISP-1, Instrument Line Flow Check Valve Operability Test, Revision 5, dated June 18, 1981, performed May 8, 1985.

-- F-ST-16I, 125 VDC Station Battery Service Discharge and Charger Performance Test, Revision 1, dated April 27, 1983, performed May 17, 1985.

-- F-ST-29D, Integrated Scram System Test, Revision 2, dated January 23, 1985, performed May 31, 1985.

-- F-ST-5N, APRM Instrument Functional Test (Refuel, Startup, Shutdown Mode), Revision 7, dated October 31, 1984, performed May 31, 1985.

The observations of the Local Leak Rate Testing (LLRT) included the post maintenance LLRT on the repaired Reactor Water Cleanup inboard containment isolation valve (12-MOV-15) and the "B" Feedwater outboard containment isolation valve (34-NRV-111B). The inspector noted that 12-MOV-15 passed the LLRT while 34-NRV-111B failed and had to be reworked. The inspector also noted that 34-NRV-111B had to be reworked several times before successfully passing an LLRT. As discussed in paragraph 6. of this report, the inspector witnessed a portion of the maintenance performed on 34-NRV-111B. Based on these observations and discussions with licensee personnel, the inspector determined that the licensee adequately performed retesting (LLRT) on repaired containment isolation valves.

No violations were identified.

6. Maintenance Observations

a. The inspector observed portions of various safety-related maintenance activities to determine that redundant components were operable, these activities did not violate the limiting conditions for operation, required administrative approvals and tagouts were obtained prior to initiating the work, approved procedures were used or the activity was within the "skills of the trade," appropriate radiological controls were properly implemented, ignition/fire prevention controls were properly implemented, and equipment was properly tested prior to returning it to service.

b. During this inspection period, the following activities were observed:

-- WR 00/21073 on the functional testing of safety related snubbers

-- WR 07/38673 on the replacement of "D" Intermediate Range Monitor dry tube.

-- WR 34/35562 on the repair of "B" Feedwater Outboard Containment Isolacion Check Valve.

-- WR 46/25455 on the repair of the "A" Emergency Service Water Pump discharge check valve.

-- WR 71/22674 on the replacement of "A" Low Pressure Coolant Injection System Battery

-- F-IMP-71.18B on the post maintenance testing of replaced HFA relays.

c. During the 1985 refueling outage in-vessel Inservice Inspection (ISI) visual examinations, the licensee identified cracks in all twelve Source Range Monitor (SRM) and Intermediate Range Monitor (IRM) instrument dry tubes. The cracks were all in the upper portion of the dry tube and were similar to those observed at other BWR's and discussed in General Electric Service Information Letter (SIL) No. 409. The indications are believed to be the result of Irradiation Assisted Stress Corrosion Cracking (IASCC). The inspector observed portions of the videotape containing the dry tube examinations. The inspector noted that the licensee individual performing the evaluations was qualified as a Level III inspector.

The dry tube ISI results were also evaluated by General Electric (GE) who concluded that the licensee could operate one additional cycle with the existing dry tubes with no adverse impact on safety. However, GE recommended that five of the dry tubes be replaced. Based on this recommendation, the licensee replaced the dry tubes at core locations 12-9, 28-33, 36-9, and 36-25 (IRM H, D, G and SRM C) which had possible indications below the bottom tube weld at the primary pressure boundary and the dry tube at 28-25 (IRM E) which had a noticeable bend at the crack location. The licensee also had to replace the dry tube at 12-33 (SRM A) after the top portion broke off when it was bumped by a double blade guide during core alterations. The inspector noted that the licensee recovered the piece which broke off.

The inspector noted that the replacement dry tubes were the same design as the original dry tubes and therefore subject to IASCC. General Electric indicated that a dry tube with materials less susceptable to IASCC would be available in the near future. The licensee tentatively plans on replacing the remaining six cracked dry tubes with ones of the new design during the next refueling outage and on replacing the six dry tubes just installed during the following outage to resolve the problem with dry tube cracking.

No violations were identified.

7. Plant Startup from Refueling

The inspectors witnessed portions of the plant startup conducted May 28-31, 1985 to verify that: the startup was performed in accordance with approved procedures; surveillance tests required to be performed prior to startup were satisfactorily completed; systems were properly aligned prior to startup; the control rod withdrawal sequence was available; and startup activities were conducted in accordance with Technical Specification requirements.

No violations were identified.

8. Shutdown Margin Demonstration

The inspector observed the Shutdown Margin (SDM) demonstration performed on May 6, 1985. The test utilized the diagonally adjacent rod method and was performed in accordance with an approved procedure. The inspector noted that the licensee terminated the test with the margin rod (22-27) at notch position 12 and the object rod (26-31), the analytically determined highest reactivity worth control rod, at position 36 after Source Range Monitor (SRM) count rate went from 60 counts per second (cps) to approximately 200,000 cps during the test. The procedure required the object rod to be fully withdrawn. Using data obtained during the test, the fuel vendor determined the SDM to be .54% $\Delta K/K$. The Technical Specifications required that the SDM for Cycle 7 be greater than .44% $\Delta K/K$. However, due to the unusual increase in SRM count rate during the test and because the calculated SDM was significantly below the SDM design value of 1.17% $\Delta K/K$, additional NRC inspections by regional specialists were conducted to evaluate the results of the tests.

As part of his followup to the SDM test, the resident inspector reviewed the completed core verification maps prepared by the licensee and noted that the final verified position of the fuel bundles was in accordance with the FitzPatrick Cycle 7 Management Report dated April 1985. The inspector noted that the verification had been performed by a Reactor Engineer and a licensed operator. A separate review of the videotapes was conducted by two Quality Control (QC) inspectors. Following the SDM test, two additional QC inspectors performed another review of the core verification videotapes. The resident inspector also viewed the core

verification videotapes and verified, for a sample of one half the core, that the fuel bundle position and orientation were in accordance with the core map. The videotapes were generally clear and the serial numbers on the fuel assemblies were adequately visible. No discrepancies were identified.

On May 25, 1985, the licensee performed a SDM demonstration using the in-sequence critical method to verify adequate SDM before the startup from the refueling outage. The inspector reviewed the test results obtained in accordance with the licensee's procedure and noted that the calculated SDM with the strongest control rod fully withdrawn was .79% $\Delta K/K$. During the plant startup on May 28, 1985, another in-sequence critical demonstration resulted in a calculated SDM of .81% $\Delta K/K$.

Based on the data reviewed, the inspector determined that the demonstrated SDM met the Technical Specification requirements. Based on a review of correspondence with the fuel vendor and on discussions with licensee personnel, the inspector noted that the deviations between the calculated SDM and the design values are probably due to calculational uncertainties. Additional details on the evaluation of the SDM demonstrations are documented in inspection reports no. 50-333/85-14 and 50-333/85-17.

9. Startup Testing-Analog Transmitter Trip System

The inspector reviewed portions of preoperational procedure no. Misc. 02A, "Preoperational Test of Analog Transmitter/Trip System for RPS and ECCS Sensor Trip Inputs (Mod. No. F1-82-53)", Revision 1, dated April 24, 1985, to verify that the procedure was properly approved and included: procedure scope and objectives; prerequisities; precautions; acceptance criteria; checkoff lists; reference to drawings and applicable procedures; provisions for recording details of the conduct of the test; provision for identification of personnel conducting the testing and evaluation of test data; and provision for quality control verification of critical steps. The inspector also verified that changes to the preoperational procedure were reviewed as required by Technical Specifications.

The inspector also witnessed portions of the testing and verified that the test was conducted in accordance with the approved procedure and that quality control verification was performed during the test. For the testing observed, the inspector noted that the test results were within the previously established acceptance criteria.

No violations were identified.

10. Followup on licensee response to General Electric Service Information Letter (SIL) No.402, Wetwell/Drywell Inerting

Based on discussions with licensee personnel and a review of Operating Experience Report No. 185, the inspector verified that the licensee evaluated the design and operation of the liquid nitrogen based inerting

system as recommended by General Electric SIL No. 402. The evaluation was performed by the Performance and Reliability Department in accordance with procedure PSO 28, "Operating Experience Feedback," and identified problems with the operation and testing of the liquid nitrogen inerting system. The inspector reviewed procedures F-OP-37, "Nitrogen Ventilation and Purge; Containment Atmosphere Dilution (CAD); Containment Vacuum Relief and Containment Differential Pressure Systems," Revision 20, and F-ST-25A, "Nitrogen System Low Temperature Simulated Automatic Isolation Functional Test," Revision 0, and determined that, in response to these findings, the licensee revised or developed procedures to verify proper operation of the nitrogen inerting system automatic isolation prior to inerting the containment and to add cautions on system monitoring if the automatic isolation valves are bypassed, such as during containment inerting directly from a nitrogen truck. The inspector also reviewed calibration data sheets dated October 2, 1984 to verify that the temperature sensors used for nitrogen system monitoring and for the automatic isolation functions were properly calibrated. The inspector verified that these sensors have been added on the calibration schedule to ensure periodic recalibration.

The inspector also noted that, during the evaluation, the licensee reviewed the portions of the liquid nitrogen system used for containment makeup during normal operations. This review noted that a modification was proposed by the architect-engineer in a letter dated November 14, 1977, to resolve a deficiency identified in Inspection Report No. 50-333/77-26 concerning the lack of low temperature isolation protection for the carbon steel nitrogen makeup lines in case of a loss of the electric heater downstream of the ambient vaporizers. Based on discussions with licensee personnel, the inspector found that this modification had not been implemented. The inspector expressed concern that no action had been taken on this problem for so long. The licensee acknowledged the inspector's concern and stated that after the refueling outage emphasis would be placed on identifying and completing these old moficiations. The inspector will review licensee progress in this area during future inspections.

The inspector reviewed the completed data sheets for procedure F-ST-39E, "Drywell to Suppression Chamber Vacuum Breaker Leak Test," performed on February 15, 1985. The purpose of this procedure is to determine the total equivalent bypass area leakage (normally expected through the vacuum breakers) between the Drywell and Suppression Pool. The test consists of maintaining a specified differential pressure (1.0 psid) between the Drywell and Suppression Pool and monitoring the rise in Suppression Pool pressure over a ten minute period. The inspector noted that the results of the test were satisfactory and there were no indications of bypass leakage. As discussed in paragraph 6. of Inspection Report No. 50-333/-84-18, the inspector has previously determined that the licensee reviewed plant data and concluded that there were no anomalies which could be indicative of Suppression Pool vent header cracks. The inspector had also previously reviewed Quality Control Inspection Report No. F84-057, which

documented the visual inspections performed on portions of the vent header, both inside and outside, including the nitrogen penetration to suppression pool shell weldment. No cracks were found during these visual inspections. The inspector noted that no ultrasonic testing of the nitrogen penetration was performed due to lack of baseline data.

Based on his review, the inspector concluded that the licensee implemented the recommendations of SIL No. 402 regarding vent header cracking. As noted above, the inspector's only concern was the licensee's failure to implement the modification needed to provide low temperature isolation protection for the nitrogen makeup lines.

11. Review of Emergency Core Cooling Systems Subject to Potential Overpressurization

The inspectors reviewed records and procedures and held discussions with licensee personnel to evaluate the design features and administrative controls that are used to minimize the potential for Emergency Core Cooling System (ECCS) overpressurization.

a. Verification of as-built isolation interfaces

The inspectors reviewed various drawings and Stone and Webster Line Designation Tables to identify those systems which contain components or piping with design pressures equal to or less than 70% of the design pressure of the primary coolant system. The inspectors noted that the High Pressure Coolant Injection (HPCI), Reactor Core Isolation Cooling (RCIC), Residual Heat Removal (including the Low Pressure Coolant Injection (LPCI) and Shutdown Cooling/Head Spray Modes), and Core Spray (CS) Systems all contain such high/low pressure interfaces. Additional details on the component configuration and the design high and low pressures can be found in attachment A to this report.

The inspectors also noted that, with respect to these systems, LPCI, CS, HPCI and RCIC all have testable check valves (valves 10-AOV-68A and B, 14-AOV-13 A and B, 23-AOV-18, and 13-AOV-22 respectively). The air operators on these valves are maintained operable and are used only during cold shutdown to verify operability of the check valve as required by Technical Specifications.

The inspectors determined that for each of the systems with a testable check valve, the air operated check valve (AOV) and the first motor operated valve (MOV) (a normally closed valve) upstream of the AOV provide isolation for the high and low pressure interface. The second MOV (a normally open valve) upstream of the AOV can also be used to provide the isolation function. For Head Spray, the isolation function is provided by a check valve and the normally closed inboard and outboard containment isolation MOV's. For Shutdown Cooling, the isolation function is provided by the normally closed inboard and outboard containment isolation MOV's.

The inspectors focused their review of surveillance and maintenance activities on the "isolation valves" identified above which normally maintain isolation for each high/low pressure interface as well as those valves which could be used to provide the isolation function.

b. Surveillance Activities

The inspectors reviewed the various surveillance test procedures listed in attachment B and held discussions with licensee personnel to determine the surveillance activities that apply to the isolation valves at each high/low pressure interface. The inspectors noted that there are several surveillance tests which are conducted to test the operation of the isolation valves for each ECCS system and RCIC. The inspector also noted that, although there is considerable overlap with ASME Section XI, the frequency of the tests are usually determined by the Technical Specifications which are more restrictive. The following is a summary of the surveillance testing performed on the isolation valves:

1. A valve operability test is performed once a month to verify the valves operate correctly when cycled from the control room. When performing this test on CS or RHR the plant may be at power or shutdown. For HPCI and RCIC the plant must be at power with steam available.

2. A system automatic actuation test is performed once a cycle by inputing simulated signals and ensuring the systems respond as appropriate. When performing this test on CS or RHR the plant must be shutdown and depressurized as the isolation valves are operated. For HPCI and RCIC the test is performed at power with the pump discharge lined up to the Condensate Storage Tank and with the inboard isolation shut and power removed.

3. A system logic functional test is performed every six months by inputing simulated signals and assuring that the system logic functions properly. The plant may be operating or shutdown when testing CS or RHR. The plant is at power when testing HPCI and RCIC. During this test the inboard isolation valve (for each system) is shut with the power removed.

4. Local Leak Rate Testing is performed on all isolation valves (except for the HPCI and RCIC testable check valve and outboard isolation valve) each refueling outage.

5. The CS, RHR, HPCI and RCIC testable check valves are cycled each cold shutdown greater than 48 hours if not done within the last 31 days.

The inspectors noted that the precautions associated with the surveillance tests are not uniform. Several of the tests contain precautions concerning opening both isolation valves simultaneously while a few others caution to ensure steps are followed in proper sequence. Some of the tests contain no precautions concerning the isolation valves. The inspectors also noted that the precautions appear to be concerned with the possibility of injecting water into the reactor vessel rather than the potential for ECCS overpressurization. However, the inspectors determined that the sequence of the test procedures (consisting of concise, specific, and identifiable steps each of which requires a verification signature on the data sheet) minimizes the potential for ECCS overpressurization.

The inspectors also noted that, in some of the tests, the interlock between the isolation valves for the low pressure ECCS systems is bypassed when simulated pressure signals are inputed. Inadvertent valve operation is prevented during these tests by removing power to the valve operators. The inspectors determined that, if a jumper is installed or an interlock bypassed, the procedure ensures that the system is returned to normal at the completion of testing.

The inspectors noted that the training for operators, with respect to the isolation valves, has basically consisted of placing industry information on valve problems (IE Information Notices, INPO SOER's etc.) in the required reading book. There is no specific training on the surveillance testing of these violation valves.

c. Maintenance Activities

The inspectors reviewed maintenance procedures, work requests and Licensee Events Reports to determine the maintenance activities and practices that apply to the isolation valves and their operators. Based on this review and discussions with licensee personnel, the inspectors noted that, in general, the licensee has not performed preventive maintenance on the isolation valves. However, the licensee indicated that a preventive maintenance program for all safety related motor operated valves (MOV's) would be implemented in the near future. This program would include items such as changing grease and checking torque switch settings on a periodic basis. With the exception of a few failures of valve motors and air operated solenoid valves, the inspectors noted that the majority of corrective maintenance on the isolation valves was due to Local Leak Rate Test failures, body to bonnet and packing leaks, problems with torque switch settings, and position indication failures. The frequency of the maintenance varied for the isolation valve involved.

Some valves required very little maintenance while others, such as the RHR, CS and RCIC testable check valves and both shutdown cooling isolation valves, required considerable maintenance.

Only one modification (other than environmental qualification upgrading) has been completed on these isolation valves. This modification was initiated to resolve recurring maintenance problems on the inboard Shutdown Cooling (SDC) isolation valve and involved rerouting the SDC line (to provide easy access to the valve for maintenance) and installing a new valve. Another recurring problem has been the failure of the disc position indication on the RHR and CS testable check valves (3 out of 4 are currently inoperable). The licensee has decided not to maintain these indicators operable due to the frequency of failures and ALARA considerations. As a result the licensee uses actuator arm and valve stem movement to verify operation during required surveillance testing. The licensee has also recently implemented procedure TOP-72, "Verification of Disk Position for CS and RHR Testable Check Valves," to verify these valves (without position indication) are shut following testing by monitoring the pressure lag across the valve during a reactor startup.

The inspectors reviewed the maintenance procedures, (listed in attachment B) used on the isolation valves and determined that they were adequate. The inspectors noted that there are separate procedures for maintenance on motor operators, pneumatic valve operators, check valves, and gate valves. Each procedure contains Quality Control inspection hold points including in the area of post maintenance testing. In general, the post maintenance testing section of each procedure requires cycling the valve several times to verify proper actuator and/or valve operation. Following valve maintenance, the licensee's Work Activity Control Procedures require the operations department to perform any additional testing to verify the valve meets the Technical Specification requirements (stroke time, leak rate, etc.) before declaring the valve operable.

Although not related to the industry problems with the isolation valves, the inspectors noted that maintenance personnel (Electricians and Mechanics) have received training on valve maintenance from valve vendors within the last two years. The inspectors noted that the amount of corrective maintenance on all safety related valves appears to be declining and the licensee attributes part of it to this training.

d. Conclusion

The inspectors noted that the licensee has one design feature which would provide early indication of an ECCS overpressurization. Specifically, the Core Spray System is annunciated and provides an alarm (Core Spray System A (B) High Pressure Valve Leakage) if the

pressure upstream of the inboard injection valve increases to 450 psig, indicating leakage by the inboard and testable check valves. In addition, the inspectors noted that, in response to industry operating experience regarding previous isolation valve problems, the licensee now has the auxiliary operator monitor and log (shiftly) HPCI and RCIC casing temperatures which would increase if there was backleakage through the isolation valves. The auxiliary operators are also required to tour (at least once per shift) the areas containing the ECCS and RCIC systems to identify and log any abnormalities some of which, such as CS and RHR system relief valve lifting or excessive pump seal leakage, may be indicative of a backleakage problem.

Based on the records reviewed and discussions with licensee personnel, the inspectors determined that there does not appear to have been any instances of actual overpressurization of the low pressure ECCS piping or components. The inspectors also determined that the maintenance and surveillance procedures reviewed appear adequate to minimize the potential for such an event.

12. Followup on a Licensee Event

On April 21, 1985, while preparing to remove a fuel support piece to allow uncoupling control rod 10-35 from the refuel floor, the licensee inadvertently lifted a fuel bundle (at location 7-36) out of the reactor core. The operators had just lowered the fuel support grapple, which was attached to the frame mounted hoist, to the upper grid. When they operated the engage button to allow the grapple to pass through the upper grid, an air leak developed which obscured the operators' vision. The grapple was raised to determine the source of the air leak. As it was being raised the air leak stopped after the operator cycled the engage and disengage buttons. When vision was restored, the operators noted that a fuel bundle had been caught on one of the grapple lock levers and had been lifted completely out of the core. The operators immediately stopped grapple motion and informed the control room. The inspector noted that, prior to and during this event, the licensee had the Standby Gas Treatment System operating and the reactor building ventilation isolated. During the event the licensee also evacuated unnecessary personnel from the Reactor Building. Additional supervisory and management personnel reported to the refuel floor.

The licensee secured the fuel bundle to the refuel bridge using a "J" hook and rope. The fuel bundle was then raised using the frame mounted hoist and transferred to the Spent Fuel Pool. The inspector noted that the bundle always remained underwater and that there was no change in the refuel floor radiation readings monitored during the event. When the bundle was lowered into a rack in the Spent Fuel Pool, it slipped off the grapple lock lever and had to be manually lowered using the "J" hook and rope. The fuel bundle was subsequently inspected by the licensee and General Electric personnel and found undamaged. The licensee counseled

all operators on this event and cautioned them to immediately stop all in-vessel operations when visual contact is lost. The event was also reviewed by the Plant Operations Review Committee who concurred and approved of the actions taken. Based on discussions with the operators and management personnel involved, and on a review of Technical Specification and Emergency Plan requirements, the inspector determined that the licensee's actions were appropriate and had no further questions regarding this event.

13. Relocation of the Emergency Operations Facility (EOF)

In a letter dated April 3, 1985, the licensee informed Region I that they planned to begin transferring equipment from the existing EOF at the Information Center to the nearly completed permanent facility at the Fulton County Airport on May 1, 1985. During the one month interval estimated for the transfer, Emergency Plan activation would necessitate that EOF functions be carried out from the Technical Support Center (TSC) until equipment was transferred back to the Visitor Center. A review of the Emergency Plan and the associated implementing procedures indicates that the TSC is formally tasked with carrying out the responsibilities of the EOF until that facility is fully activated and the TSC is relieved of those duties. The inspector verified that personnel had been designated to retrieve and set up the equipment if activation was necessary during the transition period. The reactor will be shut down until approximately May 15, 1985 to complete a refueling/maintenance outage. The plans for the transition to the new EOF were acceptable. The inspector had no further questions in this area.

14. Review of Periodic and Special Reports

Upon receipt, the inspector reviewed periodic and special reports. The review included the following: Inclusion of information required by the NRC; test results and/or supporting information consistent with design predictions and performance specifications; planned corrective action for resolution of problems, and reportability and validity of report information. The following periodic reports were reviewed:

-- March 1985 Operating Status Report, dated April 9, 1985.

-- April 1985 Operating Status Report, dated May 7, 1985.

15. Exit Interview

At periodic intervals during the course of this inspection, meetings were held with senior facility management to discuss inspection scope and findings. On June 7, 1985, the inspector met with licensee representatives (denoted in paragraph 1) and summarized the scope and findings of the inspection as they are described in this report.

Based on his review of this report, the inspector determined that this report does not contain information subject to 10 CFR 2.790 restrictions.

Attachment A

Component Configurations

The systems listed below were noted to contain components on piping with design pressures equal to or less than 70% of the design pressure of the primary coolant system.

1) Interfacing system: Core Spray
   Piping location: In
   Number of Penetrations: 2 Penetration diameter: 10 inches
   Component lineup:

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RPV-MV-AOCK-I-MOV-MOV-H/L-PRV-CK-P
    LO        NC  NO
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   Low Pressure (psig): 400
   High Pressure (psig): 1250

2) Interfacing system: Low Pressure Coolant Injection (RHR)
   Piping location: In
   Number of penetrations: 2 Penetration diameter: 24 inches
   Component lineup:

```
RCS-MV-AOCK-I-MOV-MOV-H/L-PRV-MOV-MV-CK-P
    LO         NC  NO          NO  NO
```

   Low Pressure (psig): 325
   High Pressure (psig): 1380

3) Interfacing system: Head Spray (RHR)
   Piping location: In
   Number of penetrations: 1 Penetration diameter: 4 inches
   Component lineup:

```
RPV-CK-MOV-I-MOV-H/L-PRV-CV-PRV-MOV-MV-CK-P
       NC    NC                  NO  LO
```

   Low Pressure (psig): 320
   High Pressure (psig): 1250

4) Interfacing system: Shutdown Cooling (RHR)
   Piping locations: Out
   Number of penetrations: 1 Penetration diameter: 20 inches
   Component lineup:

```
RCS-MV-MOV-I-MOV-H/L-PRV-MOV-P
    LO  NC    NC         NC
```

Low Pressure (psig): 150
High Pressure (psig): 1250

5) Interfacing system: High Pressure Coolant Injection
   Piping location: In
   Number of penetrations: 1 Penetration diameter: 14 inches
   Component lineup:

```
RPV-MV-CK-I-AOCK-MOV-MOV-P-H/L
    NO           NC  NO
```

Low Pressure (psig): 100
High Pressure (psig): 1320

6) Interfacing system: Reactor Core Isolation Cooling
   Piping location: In
   Number of penetrations: 1 Penetration diameter: 4 inches
   Component lineup:

```
RPV-MV-CK-I-AOCK-MOV-MOV-P-H/L
    NO           NC  NO
```

Low Pressure (psig): 60
High Pressure (psig): 1320

Abbreviations on this Attachment

AOCK - Air Operated Check Valve
CK - Check Valve
CV - Control Valve
H/L - High/Low Pressure Interface
I - Containment Penetration
IN - Flow Toward Reactor
LO - Locked Open
MOV - Motor Operated Valve
MV - Manual Valve
NC - Normally Closed
NO - Normally Open
OUT - Flow From Reactor
P - Pump
PRV - Pressure Relief Valve
RCS - Reactor Coolant System
RPV - Reactor Pressure Vessel

Attachment B

Procedures Reviewed

The following procedures were reviewed as part of the evaluation of the licensee's surveillance and maintenance activities on those valves which isolate primary coolant from low pressure ECCS piping and components.

1) Maintenance Procedures

- -- MP-59.3, Limitorque Motor Operators - SMB Model, Revision 3, dated November 7, 1984.
- -- MP-59.4, Maintenance Procedure for Pneumatic Valve Operators, Revision 1, dated January 10, 1985.
- -- MP-59.10, Maintenance Procedure for Non-Pressure Seal Style Gate Valves, Revision 1, dated January 16, 1985.
- -- MP-59.12, Maintenance Procedure for Non-Pressure Style Swing & Piston Check Valves, Revision 0, dated August 29, 1984.

2) Surveillance Procedures

- -- F-ST-2C, RHR MOV Valve Operability Test, Revision 13, dated November 7, 1984.
- -- F-ST-2F, LPCI and LPCI MOV Power Supply Simulated Automatic Actuation Test and LPCI Battery Service Test, Revision 15, dated April 10, 1985.
- -- F-ST-2G, RHR Isolation Valve Control Logic System Functional Test, Revision 13, dated April 17, 1985.
- -- F-ST-2H, LPCI Subsystem Logic System Functional Test, Revision 12, dated April 17, 1985.
- -- F-ST-2P, RHR Shutdown Cooling and Head Spray Simulated Automatic Isolation Test, Revision 8, dated April 10, 1985.
- -- F-ST-2S, Valve Testing - Residual Heat Removal, Revision 7, dated December 14, 1983.
- -- F-ST-3A, Core Spray/Flow Rate/Valve Operability Test, Revision 17, dated December 19, 1984.
- -- F-ST-3B, Core Spray Simulated Automatic Actuation Test, Revision 8, dated April 10, 1985.

-- F-ST-3J, Core Spray Subsystem Logic Functional Test, Revision 12, dated April 17, 1985.

-- F-ST-3M, Valve Testing - Core Spray System - Cold Shutdown Only, Revision 3, dated May 19, 1982.

-- F-ST-4A, HPCI Simulated Automatic Actuation Test, Revision 12, dated April 10, 1985.

-- F-ST-4B, HPCI Flow Rate/HPCI Pump Operability/HPCI Valve Operability Tests, Revision 19, dated January 3, 1985.

-- F-ST-4E, HPCI Subsystem Logic System Functional Test, Revision 19, dated April 10, 1985.

-- F-ST-4H, RCIC/HPCI Valve Testing, Revision 8, dated August 17, 1984.

-- F-ST-24, ISI RCIC Valve Testing, Revision 6, dated April 18, 1984.

-- F-ST-24A, RICI Pump and Valve Operability/Flow Rate Test, Revision 17, dated January 3, 1985.

-- F-ST-24E, RCIC Simulated Automatic Actuation Test, Revision 9, dated April 10, 1985.

-- F-ST-39B, Type "B" & "C" LLRT of Containment Penetrations, Revision 14, dated March 20, 1985.

-- F-ST-39J, Leak Testing of RHR and Core Spray Testable Check Valves, Revision 0, dated May 18, 1983.

3) Miscellaneous Procedures

-- TOP-72, Verification of Disk Position for CS and RHR Testable Check Valves, Revision 0, dated May 24, 1985.

-- WACP 10.1.1, Procedure for Control of Maintenance, Revision 9, dated September 28, 1984.

NRC Form 6 Rev. Oct. 80

Transaction Type
___ New Item
_x_ Modify
___ Delete

OUTSTANDING ITEMS FILE
SINGLE DOCKET ENTRY FORM

Docket Number
50-333

Doerflein	Linville
Originator	Reviewing Supervisor

Item Number Type Module # Area Resp. Action Due Date Updt/Close Date O/M/C
77-26-06 85-09-0 85-05-31
Originator Modifier/Closer
Doerflein

Description:

Item Number Type Module # Area Resp. Action Due Date Updt/Close Date O/M/C
83-04-03 85-09-0 85-05-31
Originator Modifier/Closer
Doerflein

Description:

Item Number Type Module # Area Resp. Action Due Date Updt/Close Date O/M/C
__-__-__
Originator Modifier/Closer

Description:

Item Number Type Module # Area Resp. Action Due Date Updt/Close Date O/M/C
__-__-__
Originator Modifier/Closer

Description:

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