U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No: 50-309/89-17

License No: DPR-36

Licensee: Maine Yankee Atomic Power
83 Edison Drive
Augusta, Maine 04336

Inspection At: Wiscasset, Maine

Conducted: September 1-30, 1989

Inspectors: Cornelius F. Holden, Senior Resident Inspector
Richard J. Freudenberger, Resident Inspector
Eric J. Leeds, Licensing Project Manager

Approved by: Ebe C. McCabe, Jr — 10/26/89
Ebe C. McCabe, Chief, Reactor Projects Section 3B — Date

Summary: Inspection on September 1-30, 1989 (Report Number 50-309/89-17)

Areas Inspected: Routine resident inspection of plant operations including previous inspection findings, special reports, licensee events, operational safety, maintenance, surveillance, physical security, radiation protection, and fire protection. The inspection involved 170 inspector hours including 23 backshift and 4 deep backshift hours.

Results: Observations of the repair to the Number 2 Main Feedwater Regulating Valve indicated a well coordinated and effectively implemented response (detail 6.a). Investigation into the Loose Parts Monitor indications was conducted in a timely and effective manner. Areas for improvement were noted in the implementation of the Radiation Control Program (details 6.b and 9). An anomaly identified during the routine Reactor Protective System surveillance received good follow-up (detail 7). Response to a bomb threat to the licensee's corporate offices resulted in a search of the plant and appropriate notification to the NRC (detail 8.). A review of plant modifications concluded that the licensee adequately implements 10 CFR 50.59 (detail 10).

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DETAILS

1. Persons Contacted

   Interviews and discussions were conducted with various licensee personnel, including plant operators, maintenance technicians and management staff.

2. Summary of Facility Activities

   a. The plant was at full power at the beginning of the report period. A power reduction was initiated on September 1 for routine Turbine Valve Testing. During the power reduction, difficulty with the Number 2 Steam Generator Main Feedwater Regulating Valve controller resulted in a power reduction to approximately twelve (12) percent power for repair (see detail 6.a). The valve controls were repaired and the plant was returned to full power on September 3. On September 28, a power reduction was accomplished for mussel control operations. On the following day, the plant was returned to full power, where it remained for the rest of the report period.

   b. The NRC Region I State Liaison Officer, the Division of Reactor Projects Branch 3 Chief, and the Section 3B Section Chief met with Representatives of the State of Maine, Office of Human Services to discuss interface issues on September 25, 1989.

   c. On September 27, 1989, the NRC held a meeting with Maine Yankee in the Region I office in King of Prussia, Pennsylvania, to discuss the Safety System Functional Inspection (SSFI) (Report 50-309/89-80). Attachment A to this report includes the list of attendees. Attachment B contains an outline of the items discussed. The meeting purpose was to discuss the issues and the progress Maine Yankee has made toward resolving them.

3. Review of Licensee Event Reports

   The inspector reviewed the following Licensee Event Reports (LERs) to determine that reportability requirements were fulfilled and immediate and long term corrective action was taken. The following LERs were reviewed:

88-10	Plant trip on High heater Drain Tank level
88-11	Inadvertent Safety Injections During Plant Cooldown
89-01	Plant Trip on Loss of EHC Control Power
89-02	Environmental Qualification Discrepancies Identified in Containment Cable Connector
89-03	Plant Trip Due to Inadvertent Actuation of Generator Protective Relaying

   No inadequacies were identified. The inspector found the reports to be complete and accurate.

4. Follow-up on Previous Inspection Findings

Closed - Unresolved Item (50-309/86-02-03) - The unresolved issue was the incorporation of actions specified in IE Bulletin 80-11 into engineering specifications and administrative procedures. The inspector reviewed engineering department procedures 17-21-2, "Engineering Design Change Request - Maine Yankee," 17-21-3, "Engineering Design Change Request - (YNSD)," 17-226, "Technical Evaluations," and 17-22-1, "Document Revision Procedure." Also, a recently completed modification which changed the configuration of a safety-related block wall was verified to be accurately updated in the Masonry Wall Survey Binder. No discrepancies were identified. This item is closed.

5. Operational Safety Verification

Daily, during routine facility tours, the following were checked: manning, access control, adherence to procedures and Limiting Conditions for Operations (LCOs), instrumentation, recorder traces, protective systems, control room annunciators, radiation monitors, emergency power source operability, operability of the Safety Parameter Display System (SPDS), control room logs, shift supervisor logs, and operating orders. Weekly, selected Engineered Safety Feature (ESF) trains were verified to be operable. The condition of the plant equipment, radiological controls, security and safety were assessed. Biweekly, the inspector reviewed a safety-related tagout, chemistry sample results, shift turnovers, portions of the containment isolation valve lineup, and the posting of notices to workers. Plant housekeeping and cleanliness were also evaluated.

The inspector also observed selected operations to assess safety and compliance with the NRC's regulations. The following is noteworthy.

a. Charging Pump (P-14B) Step-up Gear Failure

On September 8, the running Charging pump (P-14B), which also functions as one of the High Pressure Safety Injection (HPSI) Pumps, experienced a gradual increase in step-up gear bearing temperatures. No other abnormalities were noted with the pump's operation. When the hottest bearing reached the critical high alarm value (180 F), the pump was secured. Due to the fact that a single parameter was out of range, with no other indication of pump degradation, the licensee considered the pump to be capable of performing its safety function. The Plant Engineering Department (PED) was contacted to perform an evaluation. The pump was restarted a short time later, when PED personnel were available. Vibration data taken on the step-up gear housing were in excess of Inservice Testing (IST) operability requirements. The pump was declared inoperable and the remedial action of the Technical Specifications was entered. The spare charging pump was promptly aligned to restore the required redundancy to of the HPSI system. The inspector observed portions of the evolution in the control room, and noted that the operators' actions to identify, trend, and evaluate the equipment failure and its impact on the operability of the HPSI system were prudent. Disassembly and inspection of the step-up gear revealed that the bearings had worn excessively, however there was no damage to the gears themselves. The pump is scheduled to be returned to service by mid-October. The licensee has expanded previously scheduled plans to overhaul the spare HPSI pump (P-14S) to include an overhaul of the spare pump step-up gear as soon as the "B" pump is returned to service. The need to rebuild the "A" pump step-up gear will be evaluated after the rebuild of the "S" pump. The inspector considered the licensee's plans for inspection of similar equipment for similar failures to be appropriate.

No operational safety inadequacies were identified, and operational performance was assessed as good.

6. Plant Maintenance

The inspector observed and reviewed maintenance and problem investigation activities to verify compliance with regulations, administrative and maintenance procedures, codes and standards, proper QA/QC involvement, safety tag use, equipment alignment, jumper use, personnel qualifications, radiological controls for worker protection, retest requirements, and reportability per Technical Specifications.

Portions of the following maintenance evolutions were reviewed with no unacceptable conditions identified. Additional detailed information and inspector observations are included in the following paragraphs.

Date	Discrepancy Report Number	Description
9/1	2740-89	Main Feed Regulating Valve (FW-F-207), investigate and repair oscillations.
9/7	3476-89 3477-89 3478-89	Emergency Diesel Generator Preventive Maintenance - air start oiler check, generator brushes check, lube oil system check, integral fuel oil tank water and sediment check and air intake cleaner replacement.
9/18-22	3326-89	Loose Part Monitor System (LPMS) indication investigation.
9/14	3157-89 3158-89	PCC-M-43 and SCC-M-165 reach rod modifications.

a. <u>Main Feed Regulating Valve Failure</u>

On September 1, the licensee commenced a planned power reduction to seventy-five percent power for turbine valve surveillance and mussel control operations. As described in previous resident inspection reports, the Main Feedwater Regulating Valve (MFRV) to Steam Generator Number 2 was operating erratically. Licensee efforts to identify the cause of the oscillations had been unsuccessful. During the power reduction, the response of the number 2 MFRV became extremely limited with the controller in either automatic or manual. The operator throttled feedwater flow with the MFRV isolation valve, a motor-operated gate valve. This allowed the MFRV to go full open. The operator then used the MFRV isolation valve in conjunction with the MFRV bypass valve to control feedwater flow to the Number 2 Steam Generator.

The control room operators consulted with the Instrument and Controls (I&C) Section to ensure that the MFRV was operable to fulfill its feed train trip function as required by the Technical Specifications. The MFRV is required to close on a feed train trip signal and requires air to open against spring pressure. On a feed train trip signal, two solenoid operated valves in the instrument air supply line to the valve diaphragm vent diaphragm pressure to the atmosphere, allowing the spring to force the valve closed. This function was unaffected by the valve positioner failure, therefore Technical Specification 3.22, "Feedwater Trip System," did not apply.

Licensee management evaluated potential courses of action to allow a controlled power decrease. Options available included use of the manual handwheel of the MFRV, which would defeat the feed train trip function of the valve, and as suggested by an I&C technician troubleshooting the valve controls, the installation of a temporary modification to supply instrument air directly to the valve diaphragm. The latter approach was chosen. A temporary air pressure regulator was installed upstream of the Feedwater Trip System Solenoid-Operated Valves to provide local manual control of the air pressure to the valve diaphragm to control the MFRV position. This arrangement allowed for improved control of feedwater flow while maintaining the Feedwater Trip System operable.

With the temporary air pressure regulator installed, a controlled power reduction was commenced. At power levels below approximately fifteen percent, the MFRV bypass valve provides sufficient flow to allow the MFRV to be isolated. The valve positioner was isolated, replaced, functionally tested, and returned to service later the same day.

The inspector observed several aspects of the licensee's activities associated with this effort including: management's consideration of the potential actions available to allow repair of the valve, the briefings provided to the operators who locally operated the valve and who were in the control room, the conduct of operators both locally at the MFRV and in the control room, and the repair and functional test of the MFRV.

The inspector considered the use of the temporary modification to control the MFRV instead of taking manual handwheel control to be a positive action to enhance safe operations by reducing the amount of time the licensee would be operating the plant under the Technical Specification action statement. The inspector also observed that redundant communication equipment was available between the control room and the local manual controller, that briefings provided to the operators and I&C technicians were thorough, and that the repair work and functional testing was conducted in a cautious and timely fashion.

Licensee examination of the failed positioner determined that the failure cause was a pinhole leak in a diaphragm in the positioner. The licensee had developed a closeout plan matrix to track action items in response to the #2 MFRV oscillations. Action items to be addressed to allow completion of the closeout plan included investigation of the #1 and #3 MFRV positioners at the next shutdown, an assessment of other critical valves in the facility which have similar positioners for testing and possible replacement during the next refueling shutdown, and a survey of shared industry information to determine if this type of failure is common in the industry.

The inspector assessed the licensee's MFRV positioner repair actions as conservative and professional.

b. Indications on the Reactor Coolant System Loose Parts Monitor

To monitor for potential degradation of the thermal shield support system, which has previously generated loose parts, Maine Yankee utilizes an acoustical Loose Parts Monitoring System (LPMS). Five LPMS accelerometers are located on the reactor vessel and one is on each of the steam generators.

On September 12, routine monitoring of the LPMS indicated impacting noises on all reactor vessel and the number 1 steam generator accelerometers. The acoustic signal, although confirmed, was not of sufficient magnitude to reach the alarm setpoint. Tapes were made and sent to a consultant for evaluation. Initial evaluation indicated that the thermal shield support system was not involved, that the indication was apparently closest to the number 1 steam generator detector, that the mass was estimated to be between one and ten pounds, and that the potential loose part was likely to be in the reactor coolant system.

On September 18, three (3) impacts with sufficient magnitude to cause the LPMS to alarm were recorded. A visual inspection of the Steam Generator Number 1 (SG-1) area was then conducted, temporary accelerometers were installed on steam generator 1 and an accelerometer was moved from SG-1 to the loop 1 hot leg isolation valve on September 19, 20 and 21 respectively.

With aid from consultants, the information collected from the temporary accelerometers was evaluated. The following conclusions were made:

1) The impacts were originating from the area of the loop 1 hot leg isolation valve.

2) The impact energy was sufficiently low that little or no damage was occurring.

3) Assuming the postulated part was fully loose, its mass was approximately four (4) pounds.

Due to a potential problem identified at other facilities with excessive stress on the loop isolation valve stems, the motor-operators on the loop isolation valves at Maine Yankee had been modified during the 1988 refueling outage. In the past, the motor-operators opened the valves based on torque. The modification changed the motor-operator to cause the valve to open on limit (valve position) to reduce the stress on the stem. This change could also cause the valve to not fully backseat. That might result in impacts being generated by movement of the valve disk.

The licensee evaluated the impact of "disk flutter" on the valve internals. Based on the low energy associated with the impacts, it was concluded that the valve internals were not experiencing excessive stress and that fatigue failure was not a concern. Therefore, failure of the loop isolation valve was not deemed credible.

The preferred method of terminating a Steam Generator Tube Rupture (SGTR) is Reactor Coolant System depressurization, and the loop isolation valves are not classed as safety-related. However, the Emergency Operating Procedures (EOPs) assume the loop isolation valves are available as a secondary means of isolating a faulted steam generator in a SGTR event.

Licensee plans for future action on the LPMS indications included a monitoring program with provisions for further evaluating action levels and reactor shutdown criteria. To correct the impacting, the licensee planned an external examination of the valve during the next available shutdown. The purpose of the examination was to verify that the valve is properly backseated and there is nothing impacting the valve. An overhaul of the valve was to be scheduled for the Cycle 11/12 refueling outage in the spring of 1990.

The inspector considered the licensee's response to the identification of the part loose part to be timely, well coordinated, and thorough. Management attention provided timely review of the data by consultants and the installation of additional equipment to isolate and evaluate the consequences of the impact source. The identification and follow-up of a potential problem during the course of routine monitoring of the LPMS, prior to the indications reaching sufficient magnitude to cause alarms, was an instance of good performance.

As part of the review of the licensee's response to the potential loose part the inspector observed the containment entry to install magnetically mounted accelerometers on the number 1 steam generator. The containment entry included entries into the loop 1 area during power operation. Radiation levels in this area are routinely as high as 20-30 R/hr.

The inspector reviewed Radiation Controls Procedure 9.1.32, "Containment Loop Entry at Power," and the Radiation Work Permit (RWP) which authorized the loop entry. No discrepancies were identified. Briefings provided to the crew prior to the containment entry were thorough. However, while observing the loop entries and work performed in the annulus, the inspector made the following observations.

1) The installation of one of the magnetic accelerometers required the movement of a small section of insulation on a steam generator manway. No respiratory protection was used. No air samples were taken during this evolution. The Radiation Controls Supervisor was aware that the insulation would be moved and made the decision that air sampling or a respirator was not required.

2) The Health Physics (HP) technician assigned to cover work in the annulus area was also assigned as the security escort for one of the individuals working in the annulus area. This hampered the HP technician in monitoring other workers who were in the annulus.

3) While in the containment, an operator performed a task which was not part of the job description on the RWP that he had signed in on. In this instance the inspector concluded that the protective requirements for the task would not have been more restrictive than the RWP the individual was using.

Based on prior observation of the use of lapel air samplers for loop entries, item 1) above was considered an example of the need for the Radiological Controls Department to consistently and conservatively implement the program. Items 2) and 3) were considered examples of the need for improved support of the implementation of the Radiological Controls Program by other departments. These areas for improvement are addressed by the Radiological Controls Improvement Plan which was recently developed by the licensee. The inspector will monitor the implementation of that Plan in accordance with the routine NRC inspection program.

Overall, maintenance was assessed as good. For assessment of radiological controls, see detail 9 of this report.

7. Surveillance Testing

The inspector observed parts of tests to assess performance in accordance with approved procedures and LCOs, test results, removal and restoration of equipment, and deficiency review and resolution. The following surveillances were reviewed:

Date	Procedure Number	Title
9/1/89	3.1.3/3.1.6	Turbine Valve Testing/Excess Flow Check Valve Testing
9/14/89	3-6.2.2.9	Reactor Protection System Logic Matrix Test
9/21/89	3.17.5.1	Staff Building/Emergency Operating Facilities Ventilation Test

On September 14, during a Reactor Protective System (RPS) Logic Matrix Test, an Instrument and Controls technician noticed the AB matrix lights were energized with the selector switch in position 4. This was an abnormal indication. These lights are sometimes used as isolated circuit ground detectors and the technician believed that a ground existed in the test circuit but that it did not impact RPS operability. The condition was reported to the I&C supervisor and the remainder of the logic test was completed satisfactorily. The supervisor and the technician continued to troubleshoot the anomalous indication. They determined that the problem was associated with the bypass key switch for channel 'B' low steam generator level. The switch appeared to remain in the bypassed position. The problem also affected the AB matrix for low steam generator level. This is one of the six logic matrices, and the other five assured the safety function remained functional.

Cycling the switch cleared the condition. The licensee increased the logic matrix test frequency to once a day for a week, and then once a week for a month. There was no recurrence. The plant will soon return to monthly testing. The licensee continues to investigate replacement parts and plans to replace the switch during the next available shutdown. Until that time, the licensee plans to continue additional testing of the logic matrix to identify recurrence. The inspector reviewed Technical Specification 3.9 for minimum operable channels for the RPS and concurred that the licensee was in compliance while the switch was stuck. Also, the inspector concluded that the I&C group showed good follow-up of an anomalous indication when the results for the original logic matrix test were satisfactory.

Overall, surveillance performance was assessed as good.

8. Observations of Physical Security

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, vehicle searches and personnel identification, access control, badging, and compensatory measures when required.

On September 7, 1989, the licensee corporate offices, along with several other utilities and a local television news station in the State of Maine, received bomb threats to their offices. The threats centered around the corporate headquarters and transmission lines of these power companies. The threats indicated that the placement of the bombs was the responsibility of the "Boston Division of the New England Environmental Army." The Maine Yankee Atomic Power Station, Wiscasset, Maine was specifically exempted from the threats because of environmental considerations. The licensee evacuated and searched the corporate offices. No explosive devices were found.

The licensee's security supervisor at the Maine Yankee Atomic Power Station was notified of the threats at 11:30 a.m. on September 7, 1989 and a heightened security awareness was initiated in and around the station. Searches for explosive devices were conducted by both plant operations and security personnel in both the protected and owner-controlled areas. No explosive devices were found. The heightened security awareness posture was maintained through September 11.

The inspector observed portions of the search and the licensee's reportability determinations. Searches of all areas of the plant were conducted by teams consisting of a security officer and an operator, coordinated by a security supervisor. The Nuclear Safety Engineer made a report to the Headquarters Operations Center using the Emergency Notification System (ENS) in accordance with 10 CFR 50.72. The inspector considered the licensee's actions to search the facility, knowing that it had been specifically exempted from the threat, to be conservative.

9. Radiological Controls

Radiological controls were observed on a routine basis du[illegible] report- ing period. Areas reviewed included Organization and Ma[illegible] [illegible] radiation exposure control, and contamination con[illegible]. [illegible] 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.

The inspector witnessed a planning meeting for the low pressure safety injection pump overhaul. The meeting reviewed the problems encountered in the removal of the pump. Although the meeting participants discussed a number of problems that were encountered during the removal process, there was no firm direction on how those problems would be corrected as a result of this meeting. The lack of Engineering representation left a number of specific questions concerning the measurements of spider bearing wear open. Also, the areas of contractor training, worker and health physics coordination were discussed in general terms but no determination of specific actions were made. The inspector concluded that more specific information and action items would have been appropriate.

Overall, based on the above observations and those in Detail 6.b of this report, radiological controls performance was assessed as adequate but not significantly above the minimum requirements established by NRC regulations.

10. Periodic 10 CFR 50.59 Safety Evaluations Review

The following procedures were reviewed to determine if sufficient programmatic guidance existed for the conduct of safety evaluations pursuant to 10 CFR 50.59.

Procedure No. 0-06-4	Rev. 2	10 CFR 50.59 Determination
Procedure No. 17-21-2	Rev. 4	Engineering Design Change Request - Maine Yankee
Procedure No. 17-21-7	Rev. 2	Safety Analysis
Procedure No. 17-21-1	Rev. 2	Permanent Plant Modifications

No unacceptable conditions were noted.

The following four Engineering Design Change Request (EDCR) packages and the accompanying 10 CFR 50.59 reviews were also inspected.

EDCR 86-04 - Primary and Secondary Component Heat Exchanger Replacement.
EDCR 88-36 - Control Room Air Conditioning Cooling Upgrade
EDCR 88-45 - Containment Air Compressor System Improvements.
EDCR 88-52 - PCC/SCC Monitoring 1.97 Modifications.

No unacceptable conditions were identified. It was concluded that the 10 CFR 50.59 process with regard to engineering modifications was acceptable, and that the associated licensee performance was good.

11. State Liaison

Periodically, the resident inspectors and the onsite representative of the State of Maine discussed their findings with each other. No unacceptable plant conditions were identified.

12. Exit Interview

Meetings were periodically held with senior facility management to discuss the inspection scope and findings. The inspector continually meet with the State inspector to discuss the status of inspection findings. A summary of findings for the report period was also discussed at the conclusion of the inspection. The licensee did not identify any 10 CFR 2.790 material as being within the scope of the inspection.

ATTACHMENT A

Attendees at the Maine Yankee Safety System Functional Inspection (SSFI) meeting held in the NRC Region I Office on September 27, 1989.

Maine Yankee

Name	Title
J. Garrity	Vice President , Licensing and Engineering
J. Hebert	Manager, Plant Engineering
D. Whittier	Manager, Nuclear Engineering and Licensing
S. Nichols	Licensing Section Head

NRC

Name	Title
W. Hodges	Director, Division of Reactor Safety
J. Johnson	Chief, Reactor Projects Branch 3
E. McCabe	Chief, Reactor Projects Section 3B
G. Kelly	Chief, Technical Support Section
C. Holden	Senior Resident Inspector, DRP
J. Lyash	Project Engineer, DRP
D. Caphton	Senior Technical Reviewer, DRS
C. Woodard	Reactor Engineer, DRS
A. Giancatarino	ERC Environmental and Energy Services Company

ATTACHMENT B TO

IR 50-309/89-17

INSPECTION PRACTICES

Proactive

Resolve Concerns Before Exit

Prioritize Remaining Items

Close with Resident

TEAM INSPECTIONS

Greater Resources Required

Generate Numerous Issues

Performance Based

Closure Responsibility Unclear

RECENT TEAM INSPECTIONS

MAINTENANCE	November-December, 1988
PROCUREMENT	May-June, 1988 July, 1989
SSFI	January-February, 1989
STATION BLACKOUT	June, 1989
EOP	July, 1989

NON-RESIDENT INSPECTION HOURS

1986	768
1987	1,848
1988	2,072
1989 (To 09/01)	2,608

SALP HISTORY

1983	1.9
1984	1.78
1985	1.56
1987	1.44
1988	1.62

INTRODUCTION		G. D. Whittier
NRC Issues from SSFI		
o	Maine Yankee PRA	S. E. Nichols
	1. Safety Perspective Consistency	J. R. Hebert
	2. Cross Tie of DC Buses	S. E. Nichols
	3. Short Circuit Protection and Motor Thermal Protection	J. R. Hebert
	4. Component Cooling Heat Balance	J. R. Hebert
	5. Component Substitution Process	J. R. Hebert
o	Component Labeling	S. E. Nichols
o	Valve Maintenance Consistency	S. E. Nichols
o	Surveillance Tests for Check Valves	S. E. Nichols
o	Resolution of SSFI Matrix Items	J. R. Hebert
CLOSE		J. H. Garrity

MAINE YANKEE PRA:

Phase I	-	Internal Events	Complete
Phase II	-	CTMT Analysis	December, 1991
Phase III	-	Offsite Consequences	December, 1992

Application of PRA Results to Operations/Maintenance:

- o Prioritization of Maintenance Activities
  - oo Component Important by Itself (Significance - Moderate, Large, Very Large)
  - oo Impact of Degradation Due to Other Degraded Equipment
  - oo Relative Frequency of Problems with Components/Systems
- o Evaluation of Special Conditions
  - oo Charging Pump Anti-Pump for Example

ADMINISTRATIVE CONTROLS FOR DC BUSES:

A. Limitations on Ability to cross-Tie DC Buses:

- o Done "only in cases of immediate need...."
- o Limit "amount of time that buses are cross connected...."
- o 7 day limit whenever in Hot Standby/Power Ops (OP 1-22-2)
- o At end of seven days, shut down plant per TS 3.0.A (OP 1-22-2)

B. Operability of DC Buses 1 & 3 at Cold Shutdown:

- o Practice is to maintain at least 2 power sources available when RHR is in service.
- o "Whenever RHR Trains A or B are required to be operable, its respective DC bus, with its associated battery bank connected, will be in service." (OP 1-22-2)

OTHER ITEMS IDENTIFIED IN EXECUTIVE SUMMARY

- o Component Labeling
  - oo Applying Improved Labels to EOP Valves
- o Valve Maintenance Consistency
  - oo Two Similar Check Valves with Dissimilar PMs
  - oo No Recurring Operational Concerns Identified
  - oo In-Service Testing of Check Valves (Generic Letter 89-04)
- o Surveillance Tests for Check Valves
  - oo Generic Letter 89-04 Establishes Criteria for Check Valves
  - oo IST Program Under Revision

SAFETY PERSPECTIVE CONSISTENCY

1. Root Cause Analysis of PCC-M-43 and SCC-M-165 Failures

2. Control of Instrument Setpoints

3. Instrument Found Out of Calibration

4. Role of Maine Yankee's QA Audit Program

SHORT CIRCUIT PROTECTION AND MOTOR THERMAL PROTECTION

1. Short Circuit Protection on DC Buses

   a. Revision to DC short circuit calculations

   b. Electrical Manual

   c. Modification for "90" Refueling Outage

2. Motor Thermal Protection

   a. Evaluation Completed

   b. Setpoint Manual Being Developed

COMPONENT COOLING HEAT BALANCE

1. Complete CCW System Heat Balance

    a. Identified Prior to SSFI

    b. S&W Contracted

    c. Study Results to Date

2. Performance Testing of Heat Exchangers

    a. Identified Prior to SSFI

    b. Cooling Water System Evaluation (Generic Letter 89-13)

    c. Safeguard Heat Exchanger Performance Test Methods

    d. Improved Instrumentation

3. Shrink and Swell

    a. Not Adversely Affected Operation of Systems

    b. More Detailed/Computerized Calculations

4. Air Accumulator Capacity

    a. Reset to 92 Psig

    b. Minimum Pressure for System Performance

COMPONENT SUBSTITUTION PROCESS

1. Technical Evaluation Procedure

2. Replacement Item or Design Change