U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Report No. 84-18

Docket No. 50-271 License No. DPR-28

Licensee: Vermont Yankee Nuclear Power Corporation
RD 5, Box 169, Ferry Road
Brattleboro, Vermont 05301

Facility Name: Vermont Yankee Nuclear Power Station

Inspection at: Vernon, Vermont

Inspection Conducted: July 31 - September 18, 1984

Inspectors: William J Raymond 5 October 84
W. J. Raymond, Senior Resident Inspector date

for J. E. Tripp 10/12/84
G. W. Meyer, Project Engineer date

Stephen M. Pindale 10/12/84
S. M. Pindale, Reactor Engineer date

Approved by: J. E. Tripp 10/12/84
L. E. Tripp, Chief, Reactor Projects date
Section 3A, Projects Branch 3

Inspection Summary:
Areas Inspected: Routine, unannounced inspection on day time and backshifts by the resident and Region-based inspectors of: actions on previous inspection findings; plant power operations, including operating activities and records; Appendix R inspection followup; followup of reactor power/flow anomaly; licensee event reports; followup on plant restart issues; Conax seal replacement; post trip review procedures; procedures for review of equipment failures; and followup of a potentially generic issue (Paragraph 14). The inspection involved 195 onsite inspection hours.

Results: No violations were identified in 12 of the 13 areas inspected. One apparent violation was identified in the area of plant operations, concerning the failure to maintain core spray and residual heat removal service water system valve lineups in accordance with established procedures (paragraph 5).

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DETAILS

1.0 Persons Contacted

Interviews and discussions were conducted with members of the licensee staff and management during the report period to obtain information pertinent to the areas inspected. Inspection findings were discussed periodically with the management and supervisory personnel listed below.

Mr. J. Babbitt, Security Supervisor
Mr. B. Buteau, Reactor Engineering & Computer Supervisor
Mr. J. Desilets, Operations Supervisor
Mr. L. Bozek, Quality Assurance Engineer
Mr. R. Leach, Chemistry & Health Physics Supervisor
Mr. J. Pelletier, Plant Manager
Mr. D. Pike, Quality Assurance Manager
Mr. D. Reid, Operations Superintendent

2.0 Status of Previous Inspection Findings

2.1 (Closed) Violation 83-04-01: Failure to Maintain Secondary Containment. The corrective actions for this violation were documented in letter FVY 83-60 dated June 16, 1983. These actions were reviewed in part during Inspection 83-04 and are summarized below:

+ actions were taken immediately to restore secondary containment to a fully operational status prior to the resumption of fuel movement;

+ all personnel involved in the event were interviewed by licensee management and allowed to return to normal duties after assurance was obtained that management policies would be implemented;

+ procedure AP 0020 was changed in Revision 7 dated September 9, 1983 to specifically state that the originating department head review for a proposed jumper/lifted lead must constitute an independent evaluation of the technical adequacy of the request;

+ a sampling of previously issued jumper and lifted lead requests were reviewed to ensure that an adequate technical review had been completed. The results of this effort were reviewed by the Plant Operations Review Committee during meeting 83-37 on April 8, 1983; and,

+ the Operational Quality Assurance group conducted an independent review of design change work packages that were in progress to monitor adherence to procedural requirements. The OQA review found no evidence of a loss of management control or breakdown in the implementation of the quality assurance program at the site.

The licensee's actions for this item are considered complete and adequate. This item is closed.

2.2 (Closed) Violation 83-04-03: Responsibility for Making 50.72 Reports. Procedures AP 0010, Revision 14, and AP 0156, Revision 1, both dated January 1, 1984, clearly establish that the duty shift supervisor has the primary responsibility to determine whether 50.72 reports should be made to the NRC based on prevailing plant conditions or operational events. This item is closed.

2.3 (Closed) Follow Item 84-12-06: NRC Review of Recirculation ISI Findings. The NRC staff reviewed the findings from the 1984 inservice inspection of the reactor recirculation system piping and found that one additional cycle of operation with the current piping would be acceptable, based on the results of the ISI examinations, the corrective actions taken, and the commitments made by the licensee and documented in the NRC's Confirmatory Order dated August 28, 1984. The order is discussed further in paragraph 12 below. This item is closed.

2.4 (Closed) Follow Item 84-08-09: Staffing Changes. The Plant Manager informed the inspector on August 17, 1984 of his intent to keep the current organizational structure in the Maintenance area and to fill the personnel vacancy. Temporary reporting lines will remain in effect until the vacancy is filled. The Administrative Supervisor will now report directly to the Plant Manager. The above changes do not affect the organizational structure provided in the technical specifications. This item is closed.

2.5 (Closed) Unresolved Item 83-02-07: Notification Procedures. The inspector reviewed AP 0010, Revision 14, dated January 1, 1984 and noted that the procedure had been revised to reflect the new NRC reporting criteria in 10 CFR 50.72 and 50.73. There is no 24 hour reporting requirement in the new criteria. This item is closed.

3.0 Observations of Physical Security

Selected aspects of plant physical security were reviewed during regular and backshift hours to verify that controls were in accordance with the physical security plan and approved procedures. This review included the following security measures: guard staffing; random observations of the secondary and cental alarm stations; verification of physical barrier integrity in the protected and vital areas; verification that isolation zones were maintained; and, implementation of access controls, including identification, authorization, badging, escorting, and personnel and vehicle searches.

3.1 The actions taken by the security force during the security alert conditions that were declared on August 1 and 17, 1984 in response to protestors at the plant main gate were reviewed and found acceptable.

3.2 The inspector reviewed the status of security systems and the completion of compensatory measures for security events on August 8, 9, 13, 14, 15, 16, 17, 21, 23 and September 6 and 10, 1984. The events were attributed to power supply and/or hardware problems in the security equipment. The compensatory measures taken during the periods of equipment inoperability were acceptable. The security force implemented standing compensatory measures during the inspection period due to the excessive failures.

The inspector discussed his concerns regarding the recent hardware failure history and the guard force work hours with the Security Supervisor and with the Plant Manager during a meeting on August 17, 1984. The licensee stated that actions were being taken to expedite resolution of the hardware problems and to replace the power supplies.

This area will be reviewed further during subsequent routine inspections to verify resolution of equipment problems.

No violations were identified.

4.0 Shift Logs and Operating Records

Shift logs and operating records were reviewed to determine the status of the plant and changes in operational conditions since the last log review, and to verify that: (1) selected technical specification (TS) limits were met; (2) log entries involving abnormal conditions provided sufficient detail to communicate equipment status, correction, and restoration; (3) logs were properly completed and log reviews were conducted by the staff; and, (4) operating and special orders did not conflict with TS requirements.

The following records were reviewed periodically during the inspection:

- -- Shift Supervisor's Log
- -- Night Order Book Entries
- -- Control Point Log
- -- Radiochemistry and Chemistry Logs
- -- PRO Reports dated August 29 and 31, 1984

The Potential Reportable Occurrence (PRO) report for August 31, 1984 is discussed further in paragraph 5.0 below.

4.1 The PRO dated August 29, 1984 concerned the improper storage of radioactive liquid outside the radwaste building that was discovered by the Chemistry & Health Physics Supervisor during a tour on the morning of August 29, 1984. Three resin liners containing about 3400 gallons of waste with an activity of 6.32 mCi were moved outside during cleanup activities on the previous shift. The waste came from the radwaste building sumps that were contaminated with glycol (see paragraph 9.8 below). The liners were moved indoors by 11:00 A.M. on August 29, 1984 by licensee directive and workers were instructed to store liquid waste inside the plant.

TS 3.8.B recognizes the Waste Sample Tanks, Floor Drain Sample Tank and Waste Surge Tank as the normal (and only) outdoor storage containers for radioactive liquids, and limits the total curie content of the tanks to 3.2 Ci. The 3.2 Ci limit was not exceeded as a result of the waste in the liners. The licensee concluded that TS 3.8.B was not violated since the specification addresses the total activity of waste permanently stored outside, the liners were moved outside during actions in response to an abnormal condition, and there was no intent to increase the permanent outdoor storage capacity.

The licensee determined that this incident was not reportable since no applicable reporting criteria had been met.

No violations were identified.

5.0 Inspection Tours

Plant tours were conducted routinely during the inspection period to observe activities in progress and verify compliance with regulatory and administrative requirements. Tours of accessible plant areas included the Control Room Building, Reactor Building, Diesel Rooms, Radwaste Building, Control Point Areas, and the grounds within the Protected Area. Control room staffing was reviewed for conformance with the requirements of the Technical Specifications and AP 0036, Shift Staffing. Inspection reviews and findings completed during the tours were as described below.

5.1 Fluid Leaks and Piping Vibrations

Systems and equipment in all areas toured were observed for the existence of fluid leaks and abnormal piping vibrations. Pipe hangers and restraints installed on various piping systems were observed for proper installation and condition. No inadequacies were identified.

No violations were identified.

5.2 Plant Housekeeping and Fire Prevention

Plant housekeeping conditions, including general cleanliness and storage of materials to prevent fire hazards were observed in all areas toured for conformance with AP 0042, Plant Fire Prevention, and AP 6024, Plant Housekeeping. The inspector reviewed the installation of cable penetration fire stops between the ECCS corner rooms and the torus room. No inadequacies were identified.

The inspector reviewed the controls established by Fire Permit 84-402, which was issued on July 4, 1984 for the implementation of mechanical bypass request (MBR) 84-14. MBR 84-14 cross connected the service water and fire water systems through a 2 inch supply line downstream of fire system valve FP-321 on the Reactor Building 345 foot elevation. The fire permit and the design requirements of the fire water system were discussed with the Fire Protection Coordinator.

The fire water system is designed to maintain about 60 psi pressure in the header with one 2.5 and two 1.5 inch hose stations in operation anywhere in the system. Valve FP-321 is downstream of a 1.5 inch supply to a fire hose station located in the southeast corner of the floor. There are two other 2 inch hose stations located on the floor. The feed to the southeast hose station can be isolated by closing valve FP-313 located on the 280 foot elevation, without affecting the other two stations, which are supplied from a different header. Based on the above, the functional capability of the fire hose stations on the 345 foot elevation were not affected by the mechanical bypass.

No violations were identified.

5.3 Equipment Tagout and Controls

Tagging and controls of equipment released from service were reviewed during the inspection tours to verify equipment was controlled in accordance with AP 0140, VY Local Control Switching Rule. Controls implemented per Switching Orders 84-1091 and 1147 were reviewed. A discrepancy that occurred during the processing of tagging order 84-1147 on August 28, 1984 is discussed further in paragraph 9.0 below.

No violations were identified.

5.4 Feedwater Sparger Performance

The inspector monitored the feedwater sparger leakage detection system data and reviewed the monthly summary of feedwater sparger performance provided by the licensee in accordance with his commitment to NRC:NRR made in letter FVY 82-105. The licensee reported that, based on the leakage monitoring data received as of August 31, 1984, there were (1) no deviations in excess of 0.10 from the steady state value of normalized thermocouple readings; and (2) no failures in the 16 thermocouples initially installed on the 4 feedwater nozzles.

No violations were identified.

5.5 Safeguard System Operability

Reviews of the Residual Heat Removal, Core Spray, Residual Heat Removal Service Water, High Pressure Coolant Injection, Standby Gas Treatment and Reactor Core Isolation Cooling (RCIC) systems were conducted to verify that the systems were properly aligned and fully operational in the standby mode. Review of the above systems included the following:

- visual observation of the valve or remote position indication to verify that each accessible valve was correctly positioned.

- verification that accessible power supplies and electrical breakers were properly aligned for active components.

- visual inspection of major components for leakage, proper lubrication, cooling water supply and general condition.

No discrepancies were noted except as noted below.

5.5.1 During a valve lineup verification on August 14, 1984 with the plant operating at 15% full power (FP), the inspector found core spray (CS) valve CS-35A to be shut at 12:35 P.M. The finding was immediately reported to the Shift Supervisor and the valve was opened. CS-35A provides for pressurization of the CS discharge piping and is required to be open by OP 2123 when the CS system is required to be operable. The CS discharge piping remained pressurized with CS-35A closed, as evidenced by the observation that no changes in the CS discharge pressure occurred after the valve was opened. Thus, the requirements of technical specification 3.5.I regarding filled ECCS system discharge piping was satisfied during the period that CS-35A was closed. Based on the above, the 'A' core spray system was operable.

The last CS system valve position verification was completed on July 14, 1984 by the licensee in accordance with OP 2123, Revision 11 and CS-35A was found open at that time. CS-35A was subsequently closed with 'caution' tag 84-123 per AP 0140 on August 3, 1984 during the repair of the full flow test valve, CS-26A, which developed a motor winding fault on August 1, 1984. CS-35A was closed to prevent filling the torus from the pressurization line due to leakage past valve CS-26A. CS-26A leaked by its seat even though the valve was closed since the valve motor operator was removed for the repair. Since no work would be done under caution tag 84-123 and the control of the CS-35A position was required neither to assure worker safety nor to protect plant equipment, the use of a caution tag to control the valve instead of a white tag was acceptable.

Caution tag 84-123 was cleared on August 5, 1984 following repairs on CS-26A. The supervisory control room operator instructed the auxiliary operator (AO) to remove the caution tag and to open the valve. The AO removed the tag and used the condensate system to fill and vent the CS discharge piping, but did not open the valve. The plant was in cold shutdown on August 4, 1984 and was taken critical on August 6, 1984 to commence the startup from the refueling outage.

The failure to maintain CS-35A open was contrary to the requirements of OP 2123 and constitutes one of two examples of a violation of the requirements of technical specification 6.5.A. A second example is discussed below. This violation was discussed during a meeting with the Plant Manager on August 17, 1984 (VIO 84-18-01).

5.5.2 During a valve lineup verification at 5:05 P.M. on August 30, 1984 with the plant operating at 80% FP, the inspector identified the following mispositioned valves associated with the residual heat removal service water (RHRSW) supply to the C and D RHRSW pumps, respectively: valve RHRSW-175 XC appeared to be fully closed; and, valve RHRSW-180 XD appeared to be half open. These findings were reported to the Shift Supervisor, who went to the RHR corner rooms to open the valves. The inspector noted that valve RHRSW-175 XC was found ½ turn open and the valve was fully open at 5-3/4 turns on the hand wheel. Valve RHRSW-180 XD was found 1-1/2 turns open and 5-3/4 turns fully opened the valve. This finding was discussed with the Operations Superintendent during a meeting on August 31, 1984.

The licensee reviewed recent work activity involving the RHRSW valves and could not positively identify how the valves came to be mispositioned. Tagging orders 84-762, 768, 891, 893 and 903 were issued for work on the RHRSW lines and were cleared by July 23, 1984. A valve lineup verification was performed on July 31, 1984 per OP 2124 and valves RHRSW-175 XC and 180 XD were found open at that time. The licensee concluded it is possible that an auxiliary operator throttled the subject valves during RHRSW pump operation after July 31, 1984. The failure to maintain valves RHRSW-175 XC and 180 XD open as required by OP 2124 constitutes a second example of a violation of the requirements of technical specification 6.5.A. This item is tracked in paragraph 5.5.1 above.

A PRO was written and submitted to the Engineering Support Group to evaluate the impact on pump operability for the valves in the as found condition. The RHRSW pumps require a minimum of about 3 gpm cooling water flow to the upper bearings to assure pump operability for long term operations. The inspector reviewed the licensee's calculations for the amount of flow that would be passed by the valves in the partially open position. RHRSW-175 XC is a 3/4 inch globe valve and would have passed 2.85 GPM at 1/2 turn open. RHRSW-180 XD is a 3/4 inch gate valve and would have passed 25.3 gpm at 1-1/2 turns open. Thus, both the C and D RHRSW pumps would have had sufficient cooling to remain operable.

5.6 Radiological Controls

Radiation controls established by the licensee, including radiological surveys, condition of access control barriers, and postings within the radiation controlled area were observed for conformance with the requirements of 10 CFR 20 and AP 0503, Establishing and Posting Controlled Areas. Radiation work permit (RWP) 84-2562 was reviewed to verify conformance with procedure AP 0502, Radiation Work Permits. Work activities in progress were reviewed for conformance with the established RWP requirements. Radiation surveys were conducted by the inspector during plant tours to confirm proper posting of radiological areas. No inadequacies were identified.

5.6.1 During a tour of the plant on August 21, 1984, the inspector noted that two warning signs normally posted at the entry to the radiation control area (RCA) from the advanced offgas (AOG) catwalk were not hung in place. The signs were lying near the normal posting point and had apparently been taken down during the construction of a shelter over a portion of the catwalk. This matter was discussed with the Chemistry and Health Physics Supervisor who took actions immediately to post the entry point.

The warning signs read "Caution - Radioactive Materials - Entry by Authorized Personnel Only" and are a part of the licensee's controls to limit access to the turbine building by personnel who are not trained or badged for unescorted access in the RCA. A person who enters the RCA from the AOG catwalk does not immediately enter a radiological area as defined in AP 0503, Establishing and Posting Controlled Areas or 10 CFR Part 20, but can gain access to these areas once inside the RCA. The inspector conducted surveys and toured the areas within the turbine building and determined that a person entering the building from the AOG catwalk could not get to a radiological area defined by Part 20 that was not otherwise posted or controlled. The lack of posting on the AOG catwalk entry point was not a violation of either AP 0503 or Part 20.

The inspector had no further comment on this item for the present. Posting of entry points into the RCA will be reviewed during subsequent routine NRC inspections.

5.6.2 Instrument and Control technicians lost a check source in a grass area outside the protected area while going to the plant vent stack building on August 22, 1984. The source contained less than 1.0 uCi of Cs-137 and was used for calibration of plant radiation instrumentation. The source was found at 9:15 A.M. on August 23, 1984. The contact dose rates measured on the source by the inspector were much less than 1.0 mRem/hr, and about 3000 disintergrations per minute. The source did not pose a radiation hazard to personnel.

No violations were identified.

5.7 Jumpers and Lifted Leads (J/LL)

Implementation of J/LL Request 84-152 was reviewed to verify that controls established by AP 0020 were met, no conflicts with the Technical Specifications were created and installation/removal was in accordance with the requests.

No violations were identified.

5.8 Analyses of Process Liquids and Gases

Analysis results from samples of process liquids and gases were reviewed periodically during the inspection to verify conformance with regulatory requirements. The results of isotopic analyses of radwaste, reactor coolant, offgas and stack samples recorded in shift logs and the Plant Daily Status Report were reviewed. The conductivity of reactor vessel water increased during the period August 13-15, 1984 due to a tube leak in the main condenser. Reactor chloride levels also increased. No technical specification limits were exceeded. The inspector reviewed the chemistry sampling results to identify the presence of organic compounds in various tanks and systems following a glycol spill in the radwaste building on August 28, 1984. No technical specification limits were exceeded.

No violations were identified.

6.0 Maintenance Activities

Maintenance activity associated with the following was reviewed to verify that the requirements of AP 0021 were met and equipment return to service was proper, including the completion of operability testing.

-- MR 84-1447, Containment Sample Valve VG-26 Position Indication
-- MR 84-1657, Containment Sample Valve FSO-75B-2 Position Indication

The 'open' position indication for FSO-75B-2 was restored to an operable status during the inspection period following adjustment of the reed switch. Valve VG-26 is a containment isolation valve and its 'open' position indication was lost due to a failed reed switch. The licensee verified that the valve will operate properly as an isolation valve. The 'closed' position indication is operable. The VG-26 'open' position indication will be repaired pending receipt of a replacement reed switch. This item will be followed during a future routine inspection.

No violations were identified.

7.0 Surveillance Activities

Surveillance testing completed on August 1, 1984 in accordance with OP 4100, Revision 9, ECCS Integrated Automatic Initiation Test, was witnessed to verify that: testing was performed by qualified personnel; testing prerequisite conditions were properly established; test data demonstrated conformance with technical specification requirements and procedure acceptance criteria; and, system restoration to service was proper. The inspector noted that the RCIC 20 valve opened automatically in response to a low reactor vessel water level actuation signal.

No violations were identified.

8.0 Fire Protection Program Compensatory Measures

The licensee provided a summary in letter FVY 84-53 dated May 21, 1984 of his actions in response to the Appendix R inspection findings documented in Inspection Report 83-26. This letter also documented the compensatory measures that would be taken until certain modifications were completed. The licensee's actions were discussed in a May 24, 1984 meeting with the NRC staff and were addressed in an NRC letter to the licensee dated July 26, 1984. The July 26th letter requested that additional compensatory measures be implemented prior to plant startup from the refueling outage. The licensee responded in FVY 84-96 dated August 3, 1984 that the existing compensatory measures were considered satisfactory and no additional actions would be taken.

During a meeting with the Plant Manager on August 5, 1984 to discuss this matter, the licensee agreed to implement additional compensatory measures for the following areas: radwaste hallway; reactor building 252 foot elevation, northeast corner; and, reactor building 252 and 232 foot elevations, northeast corner. The compensatory measures would consist of placing additional fire extinguishers in the areas and assigning a roving fire watch. The compensatory measures would be placed in effect concurrent with plant startup on August 6, 1984. The above actions were considered acceptable based on the nature of the deficiencies in the identified areas, and based on the lack of transient combustibles in the areas.

The inspector reviewed the implementation of the compensatory measures on August 13 and 14, 1984, which had been in effect since August 6, 1984 based on a memorandum issued by the Fire Protection Coordinator. The administrative policy issued in the memorandum implemented the conditions agreed to in the August 5, 1984 meeting. The memorandum directed that portable fire extinguishers be placed in the following locations to supplement the extinguishers already installed in those areas:

+ RB 252 foot elevation by the elevator - could be used by a fire watch/ brigade responding to a fire in the northwest corner or northeast corner;

+ RB 252 foot elevation by the equipment hatch - for a fire in the northeast corner; and,

+ RB 252 foot elevation by the drywell access hatch - for a fire in the northeast corner.

The August 6, 1984 memorandum assigned fire watch duties to maintenance and contractor personnel who would conduct a walk through the designated areas four times per eight hour shift. A data sheet was issued to document the tours and to provide written instructions to the fire watches regarding the areas to be toured and the duties to be performed. The inspector interviewed a member of the fire watch on August 14, 1984 and found that the individual was familiar with the written instructions and the assigned duties.

During a review of the reactor building on August 13, 1984, the inspector noted that no extra fire extinguisher was installed by the drywell equipment hatch, as specified in the August 6, 1984 memorandum, but additional fire extinguishers were installed in the radwaste hallway and on the 232 foot elevation in the northwest corner. This matter was discussed with the licensee for review and resolution. The licensee was also asked to review and remove as necessary a small amount of combustible material in the northwest corner of the RB 252 foot elevation.

The licensee stated that there had been some confusion on the desired location of the extra fire extinguishers and three fire extinguishers were permanently mounted on the wall in the following locations: radwaste hallway; RB 252 foot elevation at the drywell access door; and, RB 252 foot elevation near the elevator.

The inspector had no further comment on this item at the present time. Implementation of the additional compensatory measures will be reviewed during future routine inspections.

9.0 Operational Status Reviews

Control room panels and operating logs were reviewed regularly for indications of operational problems. The operational status of standby emergency systems and equipment aligned to support routine plant operation was confirmed by direct review of control room panels. Licensed personnel were interviewed regarding existing plant conditions, facility configuration and knowledge of recent changes to procedures, as applicable. Acknowledged alarms were reviewed with licensed personnel as to cause and corrective actions being taken, where applicable. Anomalous conditions were reviewed further.

Operational status reviews were performed to verify conformance with Technical Specification limiting conditions for operation and approved procedures. The following items were noted during inspector reviews of plant operational status.

9.1 The recirculation weld leakage detection system remained operable during the inspection period, with status information available from six detectors. The system was modified during the refueling outage to monitor eight welds using six detectors in accordance with the licensee commitment to NRR, as discussed in paragraph 12 below. The system was energized continually to check the status of the detectors. No indications of recirculation system weld leakage was detected.

No violations were identified.

9.2 An inadvertent scram signal was generated during bus switching operations at 7:31 A.M. on August 1, 1984. The reactor was in the cold shutdown condition for a refueling and maintenance outage at the time.

A one-half scram signal was generated at 6:30 A.M. when the alternate 'B' RPS power supply was lost during the transfer of Bus 9 from its normal supply to Bus 8. The alternate RPS power supply and the half scram were reset. A half scram signal was again generated at 7:00 A.M. when the 'B' alternate RPS supply tripped for no apparent reason. The half scram was not reset in anticipation of subsequent bus switching operations. During the transfer of Bus 9 to its normal supply at 7:31 A.M., the 'A' RPS channel tripped due to a voltage spike induced trip on APRM channel 'C' and a full scram signal resulted.

The scram signal and RPS power supplies were reset. Shift personnel reviewed the event and determined that the cause of the trip was known and that the plant systems functioned properly. Since the trip occurred with the reactor in cold shutdown, no extensive post trip evaluation was required. The shift supervisor made a 50.72 notification to the NRC Duty Officer at 8:30 A.M.

No violations were identified.

9.3 With the plant in cold shutdown on August 4, 1984, the circuit breaker for the containment inboard isolation valve (RHR-18) tripped open at 10:30 A.M. as plant operators tried to restore shutdown cooling following completion of the cold hydrostatic test. The shutdown cooling mode of the RHR system was declared inoperable. Cooling provided by the control rod drive (CRD) system maintained reactor temperature stable at about 180 degrees F. Core decay heat was low since the plant had been shutdown since June 15, 1984. The licensee notified the NRC Duty Officer at 10:52 A.M.

Subsequent investigation by the licensee determined that the motor operator for the RHR-18 valve had a short to ground. A temporary jumper was installed on the valve position indication to remove a start inhibit on the RHR pumps. The RHR-18 valve was opened manually and shutdown cooling with the RHR system was restored at 12:58 P.M. on August 4, 1984. The RHR-18 motor was subsequently replaced with a spare at 11:30 P.M. on August 4, 1984.

The motor that failed and its associated brake were environmentally qualified and had been installed during the refueling outage under Maintenance Request 84-1299 as part of the Environmental Qualification upgrade program. The spare motor installed for RHR-18 on August 4, 1984 was qualified, but no EQ documentation exists for the motor brake. The licensee stated that both the qualified and unqualified brakes are supplied by the same manufacturer, both brakes are the same model number, and the currently installed motor brake can be shown by engineering evaluation to be qualified. This item is considered unresolved pending inspector review of the licensee's evaluation to show that the present motor and brake of RHR-18 are environmentally qualified (UNR 84-18-02).

9.4 The 'A' condensate pump failed when the control room operator tried to start it at 10:12 A.M. on August 12, 1984. The failure caused a time delayed overcurrent trip to occur on phase 'C' of the motor breaker. Subsequent investigation determined that the pump casing flange on the upper most stage of the 9 stage verticle pump had failed and the flange separated circumferentially from the casing. Since the flange of upper stage supports the weight of the rest of the pump stages, the lower stages dropped downwards.

The condensate pumps are made of cast iron and the damage caused by the August 12, 1984 failure required that the entire upper casing be replaced. Subsequent plant operation was limited initially to about 68% full power while the licensee evaluated the expected plant performance with two operating feedwater and condensate pumps. This evaluation was completed on September 8, 1984 and the licensee concluded that two condensate pumps could support full power operation. Plant load was raised to about 88% FP when the repairs to the 'A' condensate pump were completed on September 10, 1984. The 'A' pump was started at 11:06 A.M. and escalation to full power was continued with three pumps. Rated full power conditions were not achieved due to other problems, as discussed in paragraph 10 below.

No violations were identified.

9.5 During a functional test of main steam isolation valve (MSIV) 80A at 5:15 P.M. on August 8, 1984, the green 'closed' indication failed to illuminate when the valve went fully shut. The plant was operating at 25% FP at the time. Protective relays and trips associated with the valve operated as required and the valve cycled closed within the prescribed time. Plant operators considered the valve to be operable. Plant operation continued while a review of the indication failure continued. The 'open' indication on MSIV 80A was lost at 5:15 A.M. on August 9, 1984 and a ground occurred on DC Bus 1, the DC power supply for the inboard MSIV position indication. The ground cleared when the position indication circuitry for MSIV 80A was de-energized. A shutdown was begun at 4:30 P.M. on August 9, 1984 to investigate and repair the MSIV position indication circuitry. The plant entered cold shutdown on 5:05 A.M. on August 10, 1984.

Subsequent investigation determined that the indication failures were caused by failures in CONAX cables assemblies installed in the protection and position indication circuits for all 8 MSIVs during the refueling outage as part of an upgrade under the EQ program. A report under 10 CFR Part 21 was subsequently submitted regarding the CONAX failures. The Part 21 report and the assembly failure mechanism is discussed further in paragraph 13 below.

An engineering change notice (ECN) for EDCR 84-422 was prepared and implemented to remove the CONAX cable assemblies from all MSIVs. The design change package contained the engineering evaluation which justified removal of the CONAX seals. The environmental qualification of the circuits would be maintained by rearranging the old conduit runs and adding drip holes to provide for an escape path ("drip loops") for any water that might accumulate in the post accident environment. The CONAX cable assemblies were removed and plant startup was begun at 4:35 P.M. on August 11, 1984.

No violations were identified.

9.6 Reactor conductivity increased to 1.87 umhos/cm during operation at 68% FP on August 13, 1984. Main condenser chloride levels also increased. Load was reduced to 50% FP at 5:28 P.M. to investigate a suspected leak in the main condenser. A leak in the north condenser was repaired; however, load was held at about 50% FP due to little improvement in chemistry. Condenser chlorides increased to about 235 ppb and reactor conductivity increased to 2.2 umhos/cm as of 8:00 A.M. on August 14, 1984. The technical specification limits for chlorides and conductivity are 500 ppb and 5 umhos/cm, respectively. Load was reduced to 25% FP and then to hot standby conditions as the search continued. Tube leaks in the north and south water boxes of the north condenser were subsequently identified and plugged. Routine operations continued on August 16, 1984.

No violations were identified.

9.7 The 'B' blower on the steam packing exhauster (SPE) failed at 3:00 P.M. on August 16, 1984 with the plant operating at about 60% FP. The 'A' SPE blower had previously failed on August 6, 1984. The turbine building roof ventilators were shut as a precautionary measure. A small amount of leakage from the turbine shaft seals was evident by condensation on and around the seal covers. Airborne radiation levels remained below 10 CFR Part 20 limits for restricted areas at 4.0 E-9 uCi/cc.

The turbine seals were monitored periodically by operations and health physics personnel. The licensee reviewed plant operations without the SPE and concluded that continued operation was acceptable. The steam packing exhausters were subsequently repaired and returned to service. The plant continued operating at about 70% FP due to the loss of the 'A' condensate pump.

No violations were identified.

9.8 During operations at 80% FP on August 28, 1984, an error committed during the removal of tags under S&TO 84-1147 caused the 'A' glycol storage tank to overflow and about 1000 gallons of glycol/water mixture spilled to the floor of the AOG building. Some of the liquid was collected in the AOG floor drains and pumped to the equipment drain sump in the radwaste building before operators contained the spill. The glycol/water mixture became contaminated when it entered the radwaste sump.

S&TO 84-1147 was issued on August 20, 1984 to investigate a suspected leak on evaporator 'A' in the AOG system. No leak was found. An order issued by the supervisory control room operator (SCRO) at about 4:00 P.M. on August 28, 1984 resulted in valves 5026A, 5034A and 5022A being placed in the open position. The SCRO issued the order without referring to the AOG operating procedure. The required position for the valves was closed. The mispositioned valves caused solution being recirculated in the 'B' glycol storage tank to fill and overflow the 'A' tank.

The reactor continued steady state operations and there was no operational impact caused by the spill provided the glycol could be kept out of the radwaste condensate and feedwater trains. The operators took actions to confine the glycol to the radwaste sump and then to pump all collected waste into drums and resin liners for storage and subsequent disposal as radioactive and hazardous waste. Additional waste (about 4000 gallons total, including the spill) was created during subsequent operations to flush the drain lines from the AOG building to the radwaste sump to clear the lines of glycol. The inspector reviewed the licensee's actions to confine the glycol waste and then to collect, store and post it as radioactive waste. No discrepancies were identified.

No violations were identified and the inspector had no further comments regarding this event. However, the inspector noted that the above event was the third example of three recent incidents involving personnel errors that were committed during switching and tagging operations under AP 0140, as follows:

+ July 17, 1984 - the failure of contractor personnel to obtain tags prior to implementing MBR 84-14 resulted in an LCO violation;

+ August 5, 1984 - the failure of operators to open CS-35A during the release of S&TO C84-123 resulted in an improper valve lineup for the core spray system; and,

+ August 28, 1984 - the failure to refer to the AOG normal operating procedure for the release of S&TO 84-1147 resulted in a spill of glycol in the AOG building.

The control of activities under AP 0140 will be reviewed further on subsequent routine inspections.

9.9 A valve motor overload alarm was received in the control room for reactor core isolation cooling (RCIC) system valve V13-15 at 2:14 P.M. on September 8, 1984. Normal control power for the valve was lost with the valve in the open position. Plant operators declared the valve inoperable and shut the redundant isolation valve, V13-16, in the RCIC steam supply line. Preparations were made to begin alternate system testing and a 50.72 report was made to the NRC Duty Officer at 2:45 P.M.

Subsequent investigation determined that the valve motor was operable, but a GE CR120 control relay in the annunciator circuit had developed a short in the operating coil. The relay coil failure caused the fuse to blow in the control circuit. The control relay was replaced and the RCIC-15 valve was tested satisfactorily and declared operable at 8:35 P.M. on September 8, 1984.

No violations were identified.

9.10 Vacuum Breaker V16-19-11A

9.10.1 The position indication for the reactor building to torus vacuum breaker V16-19-11A was lost at 2:20 A.M. on August 25, 1984 when GE controls relay (type CR120A) in the valve annunciation circuit developed a short circuit in its operating coil. Plant operators found the valve failed open as designed since the control power to the air operator was interrupted when fuse F24/5A blew as a result of the short in the control relay. The operators declared the valve inoperable and entered the action statement of technical specification 3.7.A.5.b.

During subsequent attempts to lock the valve closed in accordance with the action statement, the operators found that the valve could not be moved without control power. (It was later determined that the air operator would have to be removed by maintenance personnel and a special tool used to operate the valve manually). Technicians completed temporary repairs under Lifted Lead (LL) 84-152 and Maintenance Request 84-1628. The inspector reviewed the circuit modifications completed under the requests.

LL 84-152 was used to isolate the failed 16A-K39 relay from the control and annunciation circuit supplied through fuse F24/5A. This action allowed the licensee to re-energize the control circuit and restore power to the solenoid for V16-19-11A. Relay K39 provides annunciation only to control room panel (CRP) 9-3 that the vacuum breaker is not fully closed. This information is a backup to the valve position indication provided on the control switch also mounted on CRP 9-3. The LL installation was completed at 4:05 A.M. and the 11A vacuum breaker was closed and declared operable at 4:10 A.M. on August 25, 1984. The K39 relay was replaced under MR 84-1628 on August 28, 1984.

The inspector reviewed the actions taken by the control room personnel and the actions required by technical specifications 3.7.A.5 and 3.7.D when a vacuum breaker is failed in the open position. These items were also discussed with the Operations Superintendent. The licensee took the position that a strict reading of the above specification would not require any operator actions nor impose any constraints on subsequent plant operations with a single vacuum breaker failed open since:

+ the open valve is serving its intended function as a vacuum breaker to protect the torus from excessive negative pressures; and,

+ the containment isolation capability for the penetration is preserved by the series check valve in the line that is normally closed. Technical specification 3.7.D allows for continued plant operations indefinitely with only one of two containment isolation valves inoperable, provided the redundant valve is secured in the isolated position.

The inspector discussed the licensee's position with NRR and NRC Region I personnel and provided the following staff position to the licensee:

+ the staff agrees that a strict reading of the VY specification would not prohibit unlimited operation with one vacuum breaker failed in the open position.

+ however, operation in the mode described above would violate the intent of the specifications which were designed to assure and preserve operability of both the containment isolation and vacuum relief functions of the valves. Additionally, preserving the containment isolation function of the valve should be given precedence over the vacuum relief function. Operation in the mode described above could lead to unacceptable consequences following an accident, given a single failure of the check valve in the penetration containing the vacuum breaker.

+ based on the above, the prompt actions taken by shift personnel on August 25, 1984 to restore the valve to an operable status was conservative, appropriate and in accord with the intent of the specifications. Should a failure similar to that on August 25, 1984 recur, the Staff would expect that the valve would be declared inoperable and it would be returned to an operable status as soon as possible.

The inspector presented this position to the Operations Superintendent during a meeting on August 30, 1984. The licensee acknowledged the inspector's comments. The inspector requested the licensee to revise technical specification 3.7.A.5 and/or 3.7.D as necessary to better reflect the intent of the requirements to preserve both the containment isolation and the vacuum relief functions of the valves. This item is open pending submittal of a technical specification change in accordance with the above and subsequent review by the NRC (IFI 84-18-03).

9.10.2 The inspector noted the following discrepancy between the apparent intended design of the torus vacuum breaker control circuitry and the actual installed wiring configuration. A note on Drawing G191175, Revision 33, states that the automatic control circuit for the vacuum breaker will override a manual signal when pressure in the torus is 0.5 psi lower than reactor building pressure. The control circuitry shown on Drawing B191301, Sheet 883, Revision 4, shows that an automatic signal to open the valve in response to a delta-P greater than 0.5 psi will not override the manual signal to close the valve if the control switch (which is spring return to auto) is held in the closed position.

This item was discussed with licensee personnel. The licensee was asked to identify any other design basis for the requirement provided on Drawing G191175. No other basis could be identified after a review of the final safety analysis report, the GE elementary drawings and the Ebasco Specifications. Drawing G191175 is considered to be in error and will be revised to delete reference to the automatic override in a future corrective update to the print. The inspector had no further comment on this item at the present time.

No violations were identified. This item is considered open pending issuance of a corrective update for Drawing G191175 to remove the identified deficiency (IFI 84-18-04).

9.11 During routine operations on September 8, 1984, the alternate power supply to the reactor protection system (RPS) from the power protection panels tripped off at 2:30 A.M. as the control room operators made a voltage adjustment on the main generator in response to a request from REMVEC. The alternate power supply was reset and returned to normal standby status.

The inspector noted that the RPS alternate power supply tripped off line on at least 4 other occasion during the inspection period, based on a review of the shift supervisors log. These events demonstrate the alternate RPS supply is very unreliable as a Class 1E power source, which can result in undesirable plant trips and transients when the alternate supply is used instead of the RPS motor generator sets. This item is open pending further review of the licensee's plans and actions to improve the reliability of the alternate power supply (IFI 84-18-05).

10.0 Core Power/Flow Anomaly

The licensee notified the inspector on September 13, 1984 of a potential core problem that was discovered during power increases on September 11, 1984.

Power ascension was in progress since September 10, 1984 when the 'A' condensate pump was returned to service following repairs. The plant had been operating at about 88% FP and 35 million lbs per hour (mpph) total core flow (73% of rated) and escalation to full power was begun after establishing the 100% rod pattern. Power increases were achieved at increments of about 1/2% per hour using recirculation flow. During power increases, the licensee routinely trends normalized core power (Pnorm), which is obtained by adjusting measured core thermal power by the difference between the existing and rated core flows, multiplied by the appropriate constant of proportionality.

Between 12 midnight and 2:00 A.M. on September 11, 1984, Pnorm decreased unexpectedly by 45 MWth (about 3% FP) while core thermal power was increased by the prescribed amount. The change in Pnorm represented an unaccounted for change in the expected core power to flow relationship. The recirculation drive temperatures increased by 2.5 and 3.0 degrees F in the A and B recirculation loops, respectively, during the same period. The core power/flow relationship departed from expected values at core flows in excess of about 42 mpph, or 87.5% of rated flow. No cause for these anomalies was immediately apparent and the licensee decreased load and held reactor power at 95% FP pending further review.

The inspector acknowledged the licensee's intended actions and requested that the NRC staff be notified prior to further escalation above 95% FP.

Engineering personnel from YAEC NSD and GE came onsite to assist in the review. Subsequent licensee review of the event over the period from September 13-15, 1984 concluded that: the measurement of core thermal power was accurate; the APRMs were accurate and responding as expected to power increases; the core thermal calculations derived from LPRM inputs were reliable and showed adequate margins (14% to 16%) to the MCPR, MFPLD and MAPLHGR core thermal limits; and, any potential core reactivity anomaly was much less than the 1% delta K/K technical specification limit.

The licensee concluded that leakage out of the shroud area was the most probable cause for the core power/flow anomaly after reviewing the core response during a power decrease to 80% FP for routine testing on September 16, 1984. The core power/flow relationship returned to expected values as core flow decreased below 42 mpph. The core delta-P corresponding to 42 mpph flow could provide the force required to lift the separator assembly. The apparent leakage out of the core shroud would alter the expected power/flow relationship by raising the annulus water temperature and thereby decrease core inlet sub-cooling. The control room operators made a 50.72 notification to the NRC Duty Officer at 11:30 A.M. on September 16, 1984 based on possible operation in an unanalyzed condition.

The licensee continued plant operations at 95% FP on September 16 and 17, 1984 while the observations from the September 16, 1984 testing were reviewed further. Administrative limits were placed at 95% FP and 40 mpph total core flow. The licensee's bases for continued plant operations were reviewed in a series of conference calls with the NRC staff on September 17, 1984. The Manager of Operations notified NRC Region I at about 5:00 P.M. on September 17, 1984 that the plant would be taken to cold shutdown to inspect the status of the reactor internals.

The licensee began a controlled plant shutdown from 95% full power (FP) at 2:00 A.M. on September 18, 1984 and the plant was taken to refueling shutdown conditions to inspect the attachment of the in-vessel steam separator/shroud head assembly to the core shroud. Plans were also made to conduct a detailed inspection of the conditions of the upper internals and the upper core area.

The initial results of the internals inspection identified that the shroud head bolts were properly orientated with respect to the hold down lugs, but were rot tight against them. As much as a 1/4 to 3/16 inch gap was observed between the T-bolt and the hold down lugs on all of the 36 bolt assemblies, using underwater tv inspection equipment. The initial inspection results also revealed evidence of apparent steam cutting on the shroud/shroud head flange near bolt #20. No definite conclusions could be reached regarding the status of the shroud to shroud head seating surface prior to further review and evaluation. A detailed inspection of the shroud and head flanges will be performed when the separator is removed from the vessel.

Licensee examination of the vessel internals in accordance with OP 2500.01 were in progress at the conclusion of the ins ction period. The scheduled examinations will be completed to document the status of various core internals and to check for abnormal conditions. Items to be inspected include: the top of fuel assemblies; annulus region around top of jet pumps; separator assembly; dryer assembly; and, the core spray piping external to the shroud.

The results of the licensee's examinations and his evaluation of the inspection results will be reviewed by the NRC staff during a subsequent inspection (IFI 84-18-06).

11.0 Licensee Event Report 84-12

The inspector reviewed licensee event report (LER) 84-12 submitted on August 16, 1984 in accordance with 10 CFR 50.73(a)(2)(ii) for the loss of secondary containment event that occurred on July 17, 1984. The report accurately described the event, with the following exceptions in the chronology:

+ contractor personnel finished work and notified control room personnel of the status of the mechanical bypass at 1:15 A.M. on July 17, 1984; and,

+ the shift supervisor arrived at the refueling floor to investigate the mechanical bypass between 3:00 and 3:30 A.M. on July 17, 1984.

The above items were discussed with the Engineering Support Supervisor, who is responsible for preparing the LERs. The discrepancies in the chronology are considered inconsequential and do not affect the NRC staff's evaluation of the event.

No violations were identified.

12.0 Plant Restart Commitments

12.1 Environmental Qualification of Electrical Equipment

On July 25, 1984, the licensee filed a request for an extension of the schedular requirements of 10 CFR 50.49(g) for replacement of components that could not be completed by the end of the 1984 refueling outage to meet the rule for the environmental qualification of electrical equipment. The 1984 outage was the second refueling outage after March 31, 1982. The licensee requested that the deadline for qualification of six motor operated valves, four solenoid valves, the local power range monitors, and the control rod position indication be extended to the end of the next refueling outage, scheduled to start in September, 1985, but in any event, no later than November 30, 1985.

The NRC staff reviewed the licensee's submittals and concluded that the request was timely, within the scope of 50.49(g), and demonstrated good cause for the schedular extension. The NRC granted the extension to complete the unfinished items by letter dated August 2, 1984.

No violations were identified.

12.2 License Amendment No. 82

The NRC issued license Amendment No. 82 by letter dated August 1, 1984 that allowed the automatic air dump system to be removed based on the permanent modifications that were made to improve the hydraulic coupling between the scram discharge and the scram instrument volumes. The licensee disabled the automatic air dump using a lifted lead request pending completion of a design change package to permanently remove the scram feature.

No violations were identified.

12.3 Recirculation Pipe Crack Related Issues

The NRC issued a confirmatory order by letter dated August 28, 1984 to confirm the actions taken by the licensee related to recirculation pipe crack issues, documented by licensee letters dated July 30 and 31, 1984. The licensee made the following commitment:

+ the licensee will operate the reactor in accordance with the revised coolant leakage limits provided in Attachment A of the Order. The revised limits were adopted and promulgated as an administrative policy at the plant.

+ the licensee installed six local leakage detectors (moisture sensitive tapes) to monitor eight uninspected 28-inch recirculation pipe welds in order to provide assurance that leakage from the welds would be promptly identified. The licensee will orally notify the NRC Project Manager within the next working day of any significant changes in the status of the detectors.

+ the licensee committed to keep the plant shutdown following the current 12-month fuel cycle until the reactor recirculation and residual heat removal system stainless steel piping are replaced.

Licensee actions to meet the above requirements will be inspected on future routine inspections.

No violations were identified.

13.0 CONAX Seals

CONAX Corporation of Buffalo, New York filed a Part 21 report with the NRC on August 31, 1984 due to a potential defect that was identified on CONAX seal assemblies. CONAX supplied 46 seal assemblies to the Vermont Yankee site and 821 to the Perry Nuclear Plant. The defects were identified as a result of the assemblies installed in the MSIV position indication circuits which failed on August 8, 1984 at Vermont Yankee.

The assemblies installed at Vermont Yankee are located in the drywell and the steam tunnel and had been subjected to about 4 days of operation at temperatures typical of normal operating conditions. The nature of the defect was a loss of electrical continuity, resulting from gradual reduction in cross sectional area of the conductors in the internal sealant area of the gland, which eventually leads to total conductor separation in some cases. The degradation appears to have been caused by the initial assembly torquing done by CONAX.

The inspector met with licensee personnel on September 6, 1984 and toured the I&C offices and the stockroom. Of 46 CONAX assemblies supplied to VY, there were 26 two conductor types and 20 eight conductor types. Sixteen eight conductor and eight two conductor assemblies were installed on the MSIVs and subsequently removed. As of September 6, 1984, all 46 of the CONAX assemblies supplied to VY were accounted for.

No violations were identified.

14.0 Potential Generic Issues

The inspector received a potentially generic issue data sheet dated July 10, 1984 from NRC Region I concerning the deletion of narrow range instrumentation requirements from the technical specifications when incorporating wide range instrumentation per NUREG 0737. The report was reviewed and the problems described therein were not applicable to the Vermont Yankee facility.

No violations were identified.

15.0 Procedures for Post Trip Reviews

NRC Generic Letter 83-28 dated July 8, 1983, requested that licensees take corrective action for problems identified during the review of the Salem Anticipated Transient Without Scram (ATWS) event. Letters FVY 83-117 dated November 7, 1983, FVY 84-116 dated September 25, 1984 and FVY 84-25 dated March 23, 1984, provide Vermont Yankee's responses to the generic letter. The inspector reviewed the licensee's actions for Item 1.1 - Post Trip Review (Program Description and Procedure).

The inspector reviewed administrative procedure AP 0154, Post-Trip Review, April 30, 1984, which provides for the following:

- -- Description of the post trip review process;
- -- Acceptance criteria for recommending restart;
- -- Responsibilities and authorities of personnel involved in the post trip review and restart decision;
- -- Requirements that a final recommendation on restart be made to the Plant Manager and that the Plant Operations Review Committee (PORC) review the post-trip reports at the next meeting; and,
- -- Format for the post-trip report, including items to be reviewed and details of the trip, with signature spaces for completion of of the review.

The inspector reviewed Post-Trip Report 84-2 for the reactor trip which occurred on April 16, 1984. The report was completed according to a preliminary version of the post-trip review procedures prior to its final approval. The inspector found the report to be a technically acceptable review.

The inspector interviewed a Duty Shift Supervisor and Duty Shift Engineers to verify that they had received training in their responsibilities under the post-trip review procedures and in interpretation of the computer outputs.

The inspector noted that there was confusion among personnel as to whether the Operations Department or PORC would maintain the file of the completed post-trip reports and that the administrative procedure did not address this area. In discussions with the inspector, Operations Department supervision stated that actions would be taken to clarify this concern.

The inspector found that the licensee's program for post-trip reviews is acceptable and meets the intent of NRC Generic Letter 83-28. Also, the inspector found the implementation of the post-trip review program for the April 16, 1984 reactor trip to be acceptable. Implementation of post-trip reviews on subsequent reactor trips will be reviewed as part of the routine inspection program.

No violations were identified.

16.0 Evaluation of Equipment Failures

The inspector reviewed the licensee's maintenance program to ensure that equipment failures are evaluated for frequency and root cause and that maintenance errors are detected, evaluated and corrected.

The licensee's review of equipment failures is accomplished primarily by two means: supervisory review of the administrative paperwork for each maintenance job and review of the equipment history records. Every maintenance job is administratively controlled and recorded on a maintenance request (MR) form. Each MR on safety-related equipment is reviewed and approved by the Maintenance Department Supervisor. His review checks to verify proper completion of the job and, based on his recollection of prior work, checks to determine whether repetitive maintenance is occurring.

The equipment history records are composed of visirecords for each component. The Visirecord contains three cards - a Preventive Maintenance (PM) Work Order, a Machine Data Card, and a Repair and PM Record. The inspector found the maintenance Visirecords to be well maintained and to contain a detailed record of all corrective and preventive maintenance work on each component for over ten years. A maintenance technician enters the information on the Visirecords. Because the repair history is readily apparent, he is able to determine whether unnecessarily repetitive maintenance work is occurring. Also, maintenance engineers periodically review the Visirecords to ensure repetitive work is not being performed and to recommend any preventive maintenance program revisions.

The inspector reviewed the implementation of the above aspects of the maintenance program. Specifically, the inspector reviewed administrative procedures AP 0200, Maintenance Program, Revision 9, April 17, 1984 and AP 0021, Maintenance Requests, Revision 11, November 18, 1982. The inspector reviewed the Visirecords for the High Pressure Coolant Injection (HPCI) system, the Service Water system, and the DC Power System for evidence of any repetitive work. The inspector found no instances of unnecessarily repetitive maintenance work which was not corrected or in the process of being revised by means of design change. To verify that corrective maintenance is consistently recorded on the Visirecords, the inspector compared the log of 1984 MRs on the HPCI system (recorded when received in the maintenance department) and the Visirecords for HPCI (recorded when completed in maintenance department) and found all MRs were consistently recorded.

In the licensee's MR system, when any work completed by the maintenance department and turned over to the plant for operation is subsequently found to be improperly repaired, the MR is resubmitted to the maintenance department. The inspector reviewed the MR log for 1983 and 1984 to find the extent of resubmitted MRs and found 4 MRs to have been resubmitted. The nature of the four resubmitted MRs was minor (non-safety related or packing leak). The inspector concluded that this was a good indication that work is done properly the first time.

The Maintenance Department Supervisor stated that improvements are underway in the maintenance program which will add equipment qualification information to the MR package, which will upgrade the description of problem root causes and track them, and which involve maintenance planners in MR job scoping before mechanics begin the work. These changes are scheduled for implementation by November, 1985. The supervisor stated that these changes are intended to further upgrade the planning and review of maintenance work so that problems such as repetitive equipment failures do not go undetected.

The inspector concluded that undetected incorrect maintenance work and unnecessarily repetitive equipment failures are infrequent in the licensee's maintenance program. Further, the inspector concluded that the well maintained Visirecord system is an asset in the licensee's review of equipment failures.

The inspector reviewed the review of work and the equipment history records in the instrumentation and controls (I&C) department. The inspector found that I&C records were also maintained on a Visirecord system and that the records were in good order. Further, all completed I&C work is reviewed by the I&C foreman and by an I&C engineer. The inspector did not review the implementation of the I&C program in detail, however, the inspector concluded the review of completed I&C work is consistent with that in the maintenance department.

No violations were identified.

17.0 Management Meetings

Preliminary inspection findings were discussed with licensee management periodically during the inspection. A summary of findings for the report period was also provided at the conclusion of the inspection and prior to report issuance.

Unresolved items are items for which further information is required to determine whether the items are acceptable or violations. An unresolved item is discussed in paragraph 9.3 of this report.