U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Report No. 84-08

Docket No. 50-271

License No. DPR-28 Priority -- Category C

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: April 3 - May 7, 1984

Inspectors: William J. Raymond 5/18/84
W. J. Raymond, Senior Resident Inspector

Robert M. Gallo 6/4/84
R. M. Gallo, Chief, Reactor Projects
Section 2A, Projects Branch 2

Inspection Summary:
Inspection on April 3 - May 7, 1984 (Report No. 50-271/84-08)

Areas Inspected: Routine, unannounced inspection on day time and backshifts by the resident inspector of: action on previous inspection findings; routine power operations; physical security; maintenance activities; surveillance testing; staffing changes; response to events; and, followup on the NRC Order dated June 27, 1983. The inspection involved 103 hours on site by the resident inspector.

Results: No violations were identified in 7 areas inspected. Three violations were identified in the area of plant operational activities, regarding the operability of the high pressure coolant injection system and the control of 125 VDC system distribution circuits; paragraph 5.

DETAILS

1. 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. W. Conway, President and Chief Executive Officer
Mr. J. Pelletier, Plant Manager
Mr. D. Reid, Operations Superintendent

2. Status of Previous Inspection Findings

a. (Closed) Unresolved Item 84-05-01: Reporting Emergency Notification System (ENS) Failures. Procedure AP 0156, Revision 1, Notification of Significant Events, dated January 1, 1984, contains a requirement that the NRC be contacted within one hour of a determination that the ENS is out of service. This requirement satisfies the IE Bulletin item. This item is closed.

b. (Closed) Violation 83-30-01: Radioactive Material Shipments. The licensee's response and corrective actions for this item were provided in letter FVY 84-18 dated March 2, 1984. Procedure AP 0504 was revised (Revision 10) on February 21, 1984 to incorporate the following items in response to the material delivered to the Beatty, Nevada burial site on September 1, 1983: (i) radiation surveys for future radwaste shipments will be made with both an RO-2A and an E520 survey instrument. The highest radiation level detected will be used for determining compliance with the transportation limits; (ii) health physics management approval will be required to ship material measured to be within 25% of the regulatory limits; and, (iii) corporate policy has been incorporated in the procedure to prohibit shipment of any packages that are measured to be greater than 90% of the federal limits.

   Procedure AP 0504 was also revised to incorporate the following items in response to the shipment of double blade guides delivered to the Pilgrim Nuclear Station on August 23, 1983: (i) information on design and manufacturing criteria for shipping packages; (ii) guidance on what constitutes 'strong' and 'tight' containers; and, (iii) a requirement that health physics management inspect questionable packages. This item is closed.

c. (Open) Violation 83-33-01: Whole Body Counting Procedures. The licensee's corrective actions were documented in letter FVY 84-15 dated March 2, 1984. The corrective actions were incomplete at the time of this inspection. A Department Instruction (DI) was prepared and submitted for review to change OP 0533, Body Burden Counting. The change to OP 0533 will require that the procedure be revised prior to future use of any whole body counting equipment other than the Applied Physics Technology system installed at the station. Further, a review is in progress by the Technical Services Superintendent to determine whether a revision to the Technical Specifications is appropriate.

This item is considered open pending issuance of the DI to OP 0533, and pending NRC review of the licensee's determination regarding the Technical Specifications.

d. (Open) Violation 83-27-01: Main Steam High Radiation Trip Setpoints. The licensee's corrective actions were described in letter FVY 83-121 dated November 23, 1983. Two of three action items were found to be complete at the time of this inspection. Test numbers 882 and 883 were added to the Master Surveillance List to provide for a once per operating cycle check of the steam line trip setpoints in accordance with OP 4511. Test number 106A was also added to the surveillance list to provide for monthly setpoint checks on the monitors.

   The third part of the licensee's corrective action concerned a commitment to revise OP 4511 to incorporate instructions on how to determine in 'normal full power background' readings on the steam line monitors for the once per operating cycle calibrations. The licensee stated in letter FVY 83-121 that OP 4511 would be completed by February 1, 1984. The revised procedure had not been issued as of May 7, 1984. The Chemistry and Health Physics Supervisor stated that the revised procedure was in the review and approval circuit and it would be issued by June 1, 1984.

   This item is considered open pending revision of OP 4511 and subsequent review by the NRC.

e. (Closed) Violation 83-29-02: Integrity of a Security Barrier. The licensee's response and corrective actions for this item were provided in letter FVY 83-125 dated December 15, 1983. The corrective actions were reviewed during a meeting with the acting Site Security Supervisor on April 23, 1984 and were found to have been satisfactorily completed. This item is closed.

f. (Closed Violation 83-27-04: SLC Valve Lineup. The corrective actions for this item were described in letter FVY 83-121 dated November 23, 1983. Licensee Event Report 83-26/3L was submitted to the NRC on October 27, 1983. This item is closed.

g. (Closed) Violation 83-27-05: Strong Motion Accelerograph Procedure Controls. The corrective actions for this item were described in letter FVY 83-121 dated November 23, 1983. OP 4396 was revised (Revision 7) on December 13, 1983 to clarify the instructions for handling the films and for running the test traces during the quarterly calibrations. A requirement was added to RP 4396 for the Department Supervisor to review and approve the completed test results, inclusive of the developed films. I&C personnel were reinstructed regarding the new procedural controls. This item is closed.

h. (Closed) Follow Item 83-17-03: HPCI/RCIC Operability Following the June 4, 1983 Vessel Depressurization. The licensee's evaluation of this event was documented in Plant Information Report 83-05 dated November 2, 1983. The licensee concluded that no damage was caused to the HPCI and RCIC steam turbines as a result of the June 4, 1983 reactor vessel blowdown. This item is closed.

i. (Closed) Follow Item 83-17-04: Vessel Transient Limits. The licensee's evaluation for this item was provided in Plant Information Report 83-05 dated November 2, 1983. Based on a review of FSAR Section 4.2.5 and other references that define reactor vessel transient limits, the licensee concluded that the June 4, 1983 depressurization did not constitute an additional transient on the vessel. No further changes to VYOPF 0145 were appropriate as a result of the June 4, 1983 event. This item is closed.

j. (Open) Violation 84-05-02: RCIC 20 Valve Logic Testing. The status of logic testing for the Reactor Core Isolation Cooling (RCIC) system valve RCIC 20 was discussed during telephone conversations with the Plant Manager on April 26 and 27, 1984. The licensee stated that further reviews of test data from the ECCS Integrated Test completed in accordance with OP 4100 in May, 1983, showed that proper operation of the RCIC 20 control circuits had been indirectly verified. However, an additional test of the RCIC 20 actuation logic will still be performed to better demonstrate proper operation of the logic circuits. The licensee stated that the logic testing would be deferred until the next shutdown due to the risk of causing a plant upset during normal power operations. The licensee plans to complete the logic testing prior to the end of the next refueling outage.

The inspector met with the Instrument and Control Department Supervisor on May 1, 1984 and reviewed the logic diagrams (CWD 191301 Sheets 1180, 1191 and 1199) for the RCIC 20 valve and the results from OP 4100 for May 31, 1983 and previous years (November 27, 1981, December 7, 1980 and October 25, 1979). The control room alarm typer edits taken during the ECCS Integrated logic tests contain a message flagging the start of the RCIC system in response to a low-low reactor vessel water level signal. The message is printed as computer point I.D. D564 - "RCIC STRT", which is derived from contacts 11 and 12 of RCIC low vessel level logic relay 13A-K2. Contacts 7 and 8 of relay 13A-K2 are used in the opening control circuit for the RCIC 20 valve. Based on these test results, proper operation of the RCIC logic was demonstrated up to the last actuating relay in the RCIC 20 automatic control circuit.

However, the following portions of the logic circuit specific to the RCIC 20 valve were not tested (verified) by the OP 4100 results: relay 13A-K2 contacts 7/8 and cable termination points AA 29 and 30 in CRP 9-30; cable termination points CC 63, 64, 65 and 66 in CRP 9-4; and, about 30 feet of cable that run between CRP 9-30 and 9-4. Technical Specification 1.0.H defines a 'Logic System Functional Test' to mean..."a test of all relays and contacts of a logic circuit from sensor to activated device to ensure all components are operable per design intent. Where possible, action will go to completion, i.e., pumps will be started and valves opened." Thus, the test results from OP 4100 do not comprise an operability test of the RCIC 20 actuation logic in a manner that meets the requirements of Technical Specification 1.0.H and 4.5.G.1.

Based on the above, the inspector noted that the licensee had identified no new information regarding the completeness of RCIC 20 logic testing beyond that already reported in Inspection Report 50-271/84-05. Based on the information presented above and reported previously, the NRC Region I staff identified no reasons to take exception to the licensee's proposed plans to complete the RCIC 20 logic test during the next refueling outage.

This item remains open pending completion of the licensee's actions to test the RCIC 20 logic circuits and subsequent review by the NRC.

k. (Open) Follow Item 84-04-04: Site Area Surveys. The licensee started a second extensive site area survey on April 30, 1984 as part of his efforts to determine whether there are any further depositions of contaminated material outside the plant radiation controlled area. The second survey was scheduled to begin after the snow had melted and is expected to be completed prior to the start of the June 16, 1984 refueling outage. The inspector met with the plant Health Physicist and the technicians who will be conducting the surveys to review the survey methodology and instrumentation. No inadequacies were identified. The survey results will be followed by the inspector during future routine inspections. This item remains open pending completion of the survey and subsequent review by the NRC.

l. (Open) Follow Item 84-04-02: Licensee Evaluation of Contamination Discovered on February 2, 1984. The licensee issued Plant Information Report 84-04 on April 6, 1984 and made a copy of the report available to the inspector for review. The report was reviewed with the Technical Services Superintendent on April 10, 1984 to determine the results of the licensee's evaluations regarding the Co-60 contamination incident and to determine the actions that will be taken to prevent recurrence.

The licensee concluded that the lump of contaminated material found near the North Warehouse was deposited sometime between July 1, 1983 and December 28, 1983 as a result of a mis-handling accident involving 55-gallon drums of aluminum oxide used in a grit-blasting decontamination operation during the 1983 refueling outage. The material most likely became deposited where it was found during the month of December, 1983. The following actions are being considered to improve the control of radioactive material: (i) evaluate and reduce as necessary the number of normal exits from the radiation controlled area (RCA); (ii) administratively require continuous health physics converage during the handling of radioactive material outside the RCA: (iii) install monitoring equipment at the Gatehouse 2 vehicle gate; (iv) conduct a detailed survey of the grounds within the site protected area on an annual basis; and, (v) evaluate the need for making the site protected area a restricted area.

The inspector had no further questions on this item for the present time. The licensee's corrective actions will be reviewed further by the NRC staff. This item is open pending finalization and completion of the proposed plans and actions.

3. 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 alarm station; 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, personnel and vehicle searches.

A security alert was declared at 4:30 P.M. on April 29, 1984 when five peaceful protestors assembled outside of the owner controlled area. The alert was relaxed after the group left the site area.

No violations were identified.

4. 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: (i) operating logs and surveillance sheets were properly completed and that selected Technical Specification limits were met; (ii) control room log entries involving abnormal conditions provided sufficient detail to communicate equipment status, correction and restoration; (iii) log book reviews were conducted by the staff; Operating and Special Orders did not conflict with Technical Specification requirements; and, (iv) jumpers (Bypasses) did not contain discrepancies with Technical Specification requirements and jumpers were properly approved prior to installation.

The following plant logs and operating records were reviewed periodically during the period of April 3 - May 7, 1984:

-- Shift Supervisor's Log
-- Night Order Book Entries
-- Jumper/Lifted Lead Log
-- Safety Related Maintenance Requests
-- Control Room Operator Logs
-- Switching Order Log
-- Shift Turnover Checklist

-- Discharge Log through 84-237
-- Radiochemistry Analysis Log
-- Equipment Status Log
-- Core Performance Log
-- Plant Information Reports 84-04 and 84-05
-- Potential Report Forms dated April 3, 9, 16, 17 and 25, 1984

No problems were identified in this area, except as noted below.

The high pressure coolant injection (HPCI) system was inoperable during the period from April 16th until April 20, 1984, as discussed in paragraph 5.0 below, when the system was returned to an operable status at 4:45 A.M. during the conduct of valve operability testing. During the April 20th testing, the HPCI turbine trip-throttle valve would not open in response to control signals, until the HPCI High Vessel Water Level Isolation Logic was reset by the control room operator. There were no entries made in the Shift Supervisor's Log for the midnight to 8:00 A.M. shift regarding the completion of the surveillance test or regarding the actions required to open the trip-throttle valve. The lack of an entry in the official shift log could have contributed to the delay until April 25, 1984 for licensee management to recognize that the HPCI system had been inoperable.

This matter was discussed with the Plant Manager on April 30, 1984. The licensee noted the inspector's comments regarding the completeness of the official shift logs and the need for complete entries regarding abnormal conditions.

No violations were identified.

5. 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.

a. 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, except as discussed below.

During a tour along the torus catwalk on April 20, 1984, the inspector noted a stem packing leak from residual heat removal (RHR) valve V10-34A.

The item was discussed with the duty Shift Supervisor and a maintenance work request was submitted to repair the leak during an outage.

No violations were identified.

b. 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.

No violations were identified.

c. 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, off-gas and stack samples recorded in shift logs and the Plant Daily Status Report were reviewed to verify that Technical Specification limits were not exceeded and that no adverse trends were apparent. Boron analysis results reported for the Standby Liquid Control System on April 6, 1984, were reviewed.

No violations were identified.

d. 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 84-205, 217, 219, 221; 232 , 268,282 and 238 were reviewed and no discrepancies were noted.

No violations were identified.

e. 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 reduced as of March 31, 1984, there were (1) no deviations in excess of 0.10 from the established constant (steady state) value of normalized temperature; and (2) no failures in the 16 thermocouples initially installed on the 4 feedwater nozzles.

No violations were identified.

f. Safeguard System Operability

Reviews of the Residual Heat Removal, Core Spray, Residual Heat Removal Service Water, High Pressure Coolant Injection, Standby Liquid Control, Standby Gas Treatment, Reactor Core Isolation Cooling and 125 VDC Distribution 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.

One violation was identified during this review. The following discrepancies were noted between the 125 VDC distribution alignment and the breaker alignment prescribed by procedure OP 2145, Normal and Emergency 125 VDC Operation, Revision 7:

(1) The breaker for vessel head spray valve RHR-33 on MCC DC-2A was required by OP 2145 to be OPEN. The breaker was found closed by the inspector at 1:00 P.M. on May 4, 1984. This finding was reported to the duty Shift Supervisor at about 1:10 P.M.

(2) The breaker for the Startup Transformer fire protection circuit on DC-2D was required by OP 2145 to be OPEN. The breaker was found closed by the inspector at about 4:30 P.M. on May 7, 1984. The condition was reported to the duty Shift Supervisor at 4:40 P.M.

The head spray line segment between downstream valve RHR-32 and the vessel head was removed as part of a 1981 design change and blank flanges were installed. The normal operational alignment for the fire protection circuit would require that its associated breaker be closed. Based on the above, there was no operational safety significance associated with the misaligned breakers. However, the above items constitute examples of a failure to follow an approved procedure (Item #1) and a failure to implement an adequate procedure (Item #2). The failure to follow approved operating procedures is contrary to the requirements of Technical Specification 6.5 (VIO 84-08-01).

g. Radiological Controls

Radiation controls established by the licensee, including: posting of radiation areas, radiological surveys, condition of step-off-pads, and disposal of protective clothing were observed for conformance with the requirements of 10 CFR 20 and AP 0503, Establishing and Posting Controlled Areas. Radiation work permits (RWPs) were reviewed to verify conformance with procedure AP 0502, Radiation Work Permits. The following RWPs were reviewed: 84-236, 313, 318 and 264.

No violations were identified.

h. Jumpers and Lifted Leads (J/LL)

Implementation of the following J/LL Requests 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: J/LL Request Nos. 84-29 through 84-35.

No violations were identified.

i. Containment Isolation

System valve lineups established to maintain containment integrity and isolation capability were reviewed on a sampling basis during inspection tours to verify conformance with the configuration specified by OP 2115, Revision 12. The review confirmed that manual valves were shut, capped and locked as required by procedure; power was available to motor operated valves and no physical obstructions would block operations; and, no leakage was evident from valves, penetrations or flanges.

No violations were identified.

j. Operational Status Reviews

Operating logs and records were reviewed for indications of operational problems and anomalous conditions were reviewed further, as required. The operational status of standby emergency systems and equipment aligned to support routine plant operation was confirmed by direct review of control room panels. The following items were reviewed to verify adherence to Technical Specification Limiting Conditions for Operation (LCOs) and approved procedures.

-- Switch and valve positions required to satisfy LCO's, where applicable, and personnel knowledge of recent changes to procedures, facility configuration and existing plant conditions.

-- Acknowledged alarms were reviewed with on shift licensed personnel as to cause and corrective actions being taken, where applicable.

-- Meter indications, recorder values, status lights, power available lights, front panel bypasses, computer printouts and comparison of redundant readings.

Operational status reviews were performed to verify conformance with the facility Technical Specifications and procedures. The following items were noted during inspector reviews of plant operational status.

(1) The recirculation weld leakage detection system (LDS) remained in a partially operable status during the inspection period, with status information available from all seven detectors. The system was energized daily to check the status of the detectors. No indication of recirculation system leakage was detected until April 10, 1984, when simultaneous leakage and trouble alarms were received on detector number 7. Plant operators increased the frequency of leakage surveillance in accordance with the Manager of Operations Directive dated January 20, 1984. Further investigation by licensee personnel and the Techmark System vendor confirmed that detector number 7 was not functioning properly and no weld leakage existed. Adjustments were made to the system circuitry to assure data could be obtained from the other six detectors.

No other anomalous conditions were noted from the six operable detectors during the remainder of the reporting period.

No violations were identified.

(2) The Toxic Gas Monitoring system inadvertently initiated at 7:20 A.M. on March 18, 1984, and was restored to a partially operable status by 10:00 P.M. on March 20, 1984 with four of five gas analyzers operable. There are no Technical Specification limiting conditions for operation for the Toxic Gas Monitoring system.

Licensee review of the system determined that a problem with the Vinyl Chloride analyzer caused the inadvertent actuation. The analyzer problem was in turn caused by a suspected faulty supply of vinyl chloride sample gas. A replacement supply of 900 ppm vinyl chloride gas was obtained and the analyzer was returned to a fully operable status on April 5, 1984.

No violations were identified.

(3) An auxiliary operator noted during operational rounds some boron precipitation on the discharge side of standby liquid control (SLC) system relief valve SR-39B. Alternate testing of the 'A' SLC pump was completed at 8:25 A.M. on May 4, 1984 to allow removal and repair of the valve. Repairs were completed under maintenance request (MR) 84-662. The SLC system was restored to a fully operable status at 3:10 P.M. on May 4, 1984 following repairs and operability testing.

No violations were identified.

(4) The control room operator noted on May 4, 1984 that voltage on 480 V Bus 89A was higher than normal. Since bus voltage was still within 10% of the required value, the bus and its uninterruptible power supply (UPS) was declared operable but degraded. Alternate system testing was started at 10:35 A.M. to allow removal of the 'A' UPS from service for repairs. Repairs were completed in accordance with MR 84-664 to replace a faulty card in the inverter gate firing module control circuitry. The A UPS was returned to service following repairs and testing at 8:27 P.M. on May 4, 1984.

The inspector had no further question on this item for the present. However, this item is considered open pending further inspector review of the maintenance history on both UPS units to determine whether any trends are apparent in the failure history (IFI 84-08-02).

(5) The licensee issued a Potential Reportable Report Form on April 9, 1984, concerning a bad cell in the 125 VDC Station Battery 'A'. The bad cell was identified as a result of quarterly surveillance testing conducted on March 30, 1984. During the surveillance test, maintenance personnel measured a float voltage of 2.03 volts and a specific gravity of 1.194 on cell #11. A battery cell is considered inoperable when float voltage is less than 2.13 and specific gravity is less than 1.19 at 77 degrees F. Surveillance on cell #11 continued daily during which a downward trend was noted. On April 9, 1984, the specific gravity decreased to 1.188 and stabilized. The Technical Specifications define a station battery to be inoperable if more than one cell is out of service.

The licensee issued MR 84-0451 to replace the faulty cell and a replacement battery was ordered. However, a decision was subsequently made to defer replacement of the cell until the plant shuts down for the refueling outage.

The inspector had no further comment on this item at the present time. This item is open pending completion of the licensee's evaluation regarding the degradation on cell #11 and replacement of the faulty cell (IFI 84-08-03).

(6) On April 16, 1984, Reactor Engineering personnel realized that the Back-Up Core Limits (BUCLE) computer program, which is used on an offsite computer on a time-share basis, had not been updated recently to revise the minimum critical power (MCPR) operating limits. The BUCLE program was used for core limits evaluation during steady state full power operations on April 13-14, 1984 and during the return to power on April 15, 1984 following a rod pattern adjustment, while the plant main computer/Vax system was out of service due to a hard disc failure. The plant computer MCPR limits had been updated from 1.29 to 1.30 earlier in the cycle in accordance with OP 0452.

Step A.9 of this procedure requires that changes be made to the BUCLE program, if applicable, when changing the plant computer software. The BUCLE program was not updated since the program was not in use at the time.

Procedure OP 4403, Core Thermal Limits Surveillance Using BUCLE, Revision 7, was in use for the period of April 13-15, 1984. A note at the start of procedure section A requires the BUCLE program be updated prior to use of the code. There are no sign-offs for the procedure step. The failure to follow OP 4403 is considered to be a licensee identified violation of the requirement of Technical Specification 6.5.

The plant computer was used for core limits evaluation for plant startup following the scram on April 16, 1984. Reactor Engineering personnel reviewed the core thermal limits for the period of April 13-15, 1984 and verified that the lowest MCPR value was 1.48. No core thermal limits were exceeded. The licensee intends to add a sign-off in OP 4403 for the requirement to update the BUCLE limits. This item will be followed on a subsequent inspection to review the revised procedure (UNR 84-08-04).

(7) The advanced offgas system automatically switched from the B recombiner to the 'A' train at 11:45 A.M. on May 7, 1984 due to an inadvertent de-energization of the 125 VAC power supply for the 'B' train instruments. The instruments de-energized when the control room operator turned the wrong power supply switch while de-energizing the 125 VDC annunciator power supply. The instruments were re-energized and the AOG process was realigned back to the 'B' train. However, the operators experienced some difficulties in re-opening the AOG suction valves, V-516 A&B, due to repeated trips of the valves caused by high pressure in a segment of the suction piping. Reactor power was reduced to 90% to avert an isolation and scram due to slowly degrading condenser vacuum conditions. The suction valves were opened and condenser vacuum returned to normal. Reactor power was returned to 100% full power.

The licensee concluded that the operator error occurred, in part, due to the arrangement of 11 power supply switches on the AOG panel. Tape was added to the panel to help delineate power supply switch groups by function and type of supply.

No violations were identified.

(8) During routine power operations on April 19, 1984, the 'B' recirculation pump tripped off line at 2:30 A.M. No apparent reason was identified for the trip. The pump was restarted at 3:30 A.M. and power escalation resumed. The 'B' pump tripped off line again at 5:07 A.M. for no apparent reason. Plant power

was held at 65% full power using the 'A' recirculation pump while maintenance personnel investigated the pump protection and control circuitry. No definite problem was identified. The 'B' pump was restarted at 1:20 P.M. on April 19, 1984. Maintenance personnel subsequently identified and replaced a bad capacitor in the 'B' motor generator set phase ground detection circuitry; however, no definitive cause for the previous trips was identified. Operation with both pumps continued for the remainder of the reporting period without further incident.

Technical Specification 3.6.G permits reactor operation with a single recirculation loop for 24 hours. The inspector reviewed the surveillance actions taken in response to the pump trips for conformance with Technical Specification 3.6.A.4 and OP 2110. Reactor Engineering personnel performed surveillance on the average power range monitor (APRM) channels during single loop operations to assure that APRM trip and rod block setpoints were proper.

No violations were identified.

(9) The inspector met with the Reactor Engineering and Computer Supervisor on April 24, 1984 to discuss a problem that was identified by the licensee regarding new fuel receipt inspections that were completed in March, 1984. New fuel bundles were wiped down with acetone during receipt inspection in accordance with OP 1401 prior to storage in the spent fuel pool. Sixteen fuel bundles were cleaned on March 28, 1984 with a batch of acetone that was later determined to be suspect due to a 'peculiar' odor associated with it. A sample of the material was sent offsite for chemical analysis to determine its composition. The preliminary results of the analysis were reported to the licensee on April 20, 1984. The batch of acetone in use on March 28, 1984 was found to be contaminated with acetic acid at a level of 4500 ppm. Trace contaminants in acetone are normally in the range of 100 ppm.

The application of contaminated acetone on the 16 fuel bundles was reviewed by the licensee in conjunction with the fuel supplier. The licensee concluded that no adverse affects on the fuel would occur and the fuel will be used for cycle 11 operations.

The licensee stated that the acetone used on March 28, 1984 was obtained from the station stockroom. A review of the controls provided for acetone used for fuel inspections was initiated. The controls in OP 1401 will be upgraded to assure that only specially designated sources of acetone will be used in the future.

The inspector had no further comments on this item at the present time. This area will be reviewed on a subsequent inspection pending completion of the licensee's reviews and corrective actions (IFI 84-08-05).

(10) Routine surveillance on the residual heat removal (RHR) system was completed on April 3, 1984 and the 'A' RHR loop was left in service for torus cooling. When the Torus Spray Test Discharge valve, RHR-39A, was closed at 5:50 A.M. to secure from torus cooling, the valve motor breaker tripped and a motor thermal overload alarm was received in the control room. Plant Personnel inspected the valve and found that the motor circuit breaker on MCC 9B had tripped open on short circuit protection, the motor windings had burned out, the motor operator housing was cracked open and the valve operator was jammed with the valve in the closed position.

The 'A' loop of torus cooling was declared inoperable and alternate system surveillance testing was begun. RHR valves 38A and 34A were confirmed to be closed at 8:00 A.M. to assure compliance with the containment isolation provisions of Technical Specification Table 4.7.2.b. A replacement motor and operator was installed for RHR-39A and the system was returned to an operable status at 7:30 P.M. on April 5, 1984 following operability testing.

The inspector reviewed the circumstances associated with the valve failure, the actions taken to replace the valve motor operator, the surveillance testing completed while the RHR-39A valve was out of service and the testing completed on April 5, 1984 to verify proper stroke time and leak tightness of RHR-39A. Maintenance work completed on valve RHR-39A per MR 84-469 is discussed further in paragraph 7 below.

No violations were identified.

(11) The licensee notified the NRC staff on April 25, 1984 of a degraded HPCI system actuation logic condition that existed at the plant from April 16-20, 1984. The reactor operated at power during this time. The licensee's telephone notifications of the event included a report to the NRC Duty Officer in accordance with the requirements of 10 CFR 50.72.b.2.(iii). The resident inspector reviewed the circumstances associated with the incident upon his return to the site on April 30, 1984. The inspection included a review of applicable procedures and drawings, and interviews with members of Operations Shift Crew C and D. The findings from this review were as summarized below.

The plant scrammed at 7:42 A.M. on April 16, 1984 due to MSIV closure caused by high main steam line flow when the 80-C MSIV failed shut on a 10% closure test. Operations Crew D was on shift when the scram occurred. The operators used OP 3100, Reactor Scram Emergency Procedure, Revision 12, to recover from the scram and to stabilize the plant. The transient can be characterized as a reactor scram caused by an MSIV isolation at full power, which resulted in lifting three of four safety relief valves, and was complicated by indications that the 'D' safety relief valve had failed to reseat.

A high reactor water level condition occurred during the scram recovery which caused a sealed-in trip signal to be imposed on the HPCI system trip throttle valve. When vessel water level subsequently decreased below the 177 inch high level trip setting, the control room annunciators which alarmed the high level condition cleared, but the HPCI high level isolation logic remained in the tripped condition, as designed. It is normal for the logic to remain in the tripped condition until the logic reset pushbuttons are depressed on control room panel CRP 9-3. However, the logic channel provides no indications to the reactor operator regarding its tripped status. With the HPCI high water level isolation logic in the tripped condition, the HPCI system will start in response to a low reactor vessel level actuation signal, but not in response to a high drywell pressure actuation signal.

The high water level isolation logic was not reset as part of the scram recovery actions or during subsequent actions to stabilize the plant. Step 10 of OP 3100 requires as a subsequent action following a scram that the alarm typer printout be reviewed for alarm conditions that may have cleared during the transient. The alarm typer printout for the period following the April 16, 1984 scram shows that the high vessel water level condition occurred at 7:43 A.M. and subsequently cleared at 7:51 A.M. This combination of messages, coupled with a lack of direct indication of the logic tripped status, may have contributed to the failure of the Shift Crew D to reset the isolation logic. Following plant stabilization from the scram, Shift Crew D oversaw activities to repair MSIV 80-C and the position indication for safety relief valve 'D', and to begin a verification that prerequisite conditions were met to restart the plant.

A plant startup was commenced by Operations Crew C at 6:15 P.M. on April 16, 1984. Two procedures governed the plant restart effort: OP 0100, Reactor Startup to Criticality, Revision 14; and, OP 0101, Reactor and Generation System Heatup to Low Power, Revision 13. Upon assuming shift duties, the Supervisory Control Room Operator (SCRO) took responsibility for completing the reactor startup in accordance with OP 0100 and 0101. Since the plant was maintained in the hot shutdown condition since the scram, the prerequisite and sequential steps of both procedures were partially satisfied when the procedures were entered. The SCRO reviewed the procedure requirements against the then existing plant conditions and determined that Step 8 of OP 0100 was the appropriate entry point to resume startup verifications. Step 4 of OP 0100 lists 13 safeguard system logic 'reset' pushbuttons on CRP 9-3 and 9-4, including the HPCI high water level reset pushbutton, that must be depressed as part of the startup procedure. The SCRO stated during an interview on May 2, 1984 that the requirements of this step were assumed to have been satisfied since all safeguard systems were listed as 'operable' during the shift turnover at 3:30 P.M. on April 16, 1984. The reactor was taken critical at 8:38 P.M. and power operations resumed.

Operations Shift Crew C performed the monthly HPCI valve operability test at about 4:45 A.M. on April 20, 1984. During the valve test, it was noted that the trip throttle valve did not open as expected when the auxiliary oil pump was started. After a second unsuccessful attempt was made to open the trip throttle valve with the same results, the SCRO recalled the high water level condition following the April 16, 1984 scram and pushed the HPCI high water level reset pushbutton. This action cleared the high level isolation signal and the trip throttle valve went open. The three licensed operators and the shift engineer discussed the events that had just transpired and concluded that the HPCI system operability had not been affected by the high level isolation signal, since the high level isolation signal would be overridden by a low vessel level signal to provide for HPCI operation under accident conditions. No consideration was given to the high drywell pressure portion of the actuation logic during the discussion. There was no reference made to system manuals or logic diagrams. No entry was made in the Shift Supervisor's log regarding completion of the surveillance test nor the difficulties encountered in opening the trip throttle valve.

The Crew C Shift Engineer attended a training class during the week of April 30, 1984, and raised questions regarding actions taken on April 20, 1984 to open the trip throttle valve. It was realized during the ensuing review of the test sequence and the system logic diagrams that a problem could have existed with operability of the HPCI system during the time from the trip on April 16, 1984 until April 20, 1984 when the logic was reset. The Shift Engineer discussed his concerns with the Operations Engineer on April 24, 1984 and with the Crew C Shift Supervisor at 7:30 A.M. on April 25, 1984. The HPCI operability question was presented to the Operations Supervisor for evaluation and guidance on April 25, 1984. It was determined through discussions with the I&C and Engineering Support Supervisors that from the time of the high level trip until the logic was reset, the High Drywell initiation circuitry of HPCI was invalidated (locked out).

Technical Specification Table 3.2.1 lists the HPCI actuation instrument channels that must be operable when the HPCI system is required to be operable by Technical Specification 3.5 (whenever irradiated fuel is in the reactor and pressure is greater than 150 psig). The instruments include trip functions based on low vessel water level and high drywell pressure. Note 5 of the Table states that if the minimum number of operable instrument channels is not available, which was the case from April 16-20, 1984 when the high drywell pressure portion of the actuation logic was locked out, then the system is considered inoperable and the conditions of Technical Specification 3.5 apply. Technical Specification 3.5 allows for continued reactor operation for up to 7 days with the HPCI system inoperable, provided all active components of the LPCI, CS, ADS and RCIC systems are operable. Technical Specification 4.5.E.2, which is part of the Technical Specification 3.5.E.2 limiting condition for operation, requires that the alternate systems be tested immediately and daily thereafter whenever the HPCI system is made or found to be inoperable for any reason.

Although the licensee did not know of the degraded HPCI condition from April 16-20, 1934, sufficient plant status information existed such that the condition could have been known if the information were reviewed (the high water level condition was permanently recorded in the control room by the alarm typer and the narrow range level recorder on CRP 9-5). Additionally, sufficient procedure controls existed to have precluded the degraded condition, had they been followed. The failure to have the HPCI system operable from April 16-20, 1984 is a violation of Technical Specification 3.5.E (VIO 84-08-06).

The failure to follow the requirements of procedure OP 0100 on April 16, 1984 to reset the HPCI high water isolation reset pushbutton constituted a violation of the requirements of Technical Specification 6.5 (VIO 84-08-07).

The inspector noted, based on a review of Sections 6.0, 7.0 and 14.0 of the Final Safety Analysis Report, that no credit is taken for the high drywell pressure portion of the actuation logic as the primary HPCI actuation signal in response to any analyzed accident.

During meetings with the Plant Manager, the Operations Superintendent and the Operations Supervisor, the inspector noted that certain corrective actions were in progress, which included: (i) making a temporary change to OP 0100 to make it mandatory (via addition of a sign-off requirement) that the safeguard reset pushbuttons be depressed as a prerequisite for reactor startup;

(ii) review of the installed hardware to determine if changes should be made to provide indication of the safeguard system logic status; and, (iii) adding the HPCI operability event as a review item in the operator requalification lecture series, along with a review of the HPCI actuation logic design. The latter item will be covered in the requalification lecture series starting in May, 1984. The licensee's corrective actions will be reviewed further on a subsequent inspection.

6. Surveillance Activities

The inspector reviewed portions of the following tests to verify that testing was performed by qualified personnel; test data demonstrated conformance with Technical Specification requirements; and, system restoration to service was proper.

-- OP 4114.01, SLC Pump Capacity Data Sheet, May 4, 1984
-- OP 4123.01, Core Spray System Surveillance, May 4, 1984
-- OP 4124.04, RHR Pump Operability Data Sheets, April 3, 1984
-- OP 4124.06, RHR Pump and Valve Operability, April 3, 1984
-- OP 4124.01, RHR Valve Operability Data Sheet, April 3, 1984
-- OP 4113.01, MSIV Partial Closure Test, April 2, 1984
-- OP 4401.01, Core Thermal Limits Evaluation, April 4, 1984
-- OP 4121.05, RCIC Operability Test, April 20, 1984
-- OP 4120.01, HPCI Operability Test, April 20, 1984
-- OP 4115.04, Drywell/Torus Vacuum Breaker Test, April 16, 1984
-- OP 4113.01, MSIV Partial Closure Test, April 16, 1984

No violations were identified.

7. Maintenance Activities

The maintenance request log was reviewed to determine the scope and nature of work done on safety related equipment. The review confirmed: the repair of safety related equipment received priority attention; Technical Specification limiting conditions for operation (LCOs) were met while components were out of service; and, performance of alternate safety related systems was not impaired.

Maintenance activity associated with the following was reviewed to verify that delay of work was acceptable for those items deferred to plant shutdown, and for those items where work was completed, that the requirements of AP 0021 were met, qualified replacement parts were used, administrative approvals and tagouts were proper and equipment return to service was proper, including the completion of operability testing.

-- MR 84-469, RHR Valve 39A, April 3, 1984
-- MR 84-451, Cell #11 of Station Battery 'A', April 23, 1984
-- MR 84-480, Service Water Valve 55B, April 5, 1984
-- MR 84-521, Service Water Pump 'D' Packing Leak, April 20, 1984

-- MR 84-536, Moisture Sensitive Detector #7, April 23, 1984
-- MR 84-564, MSIV 80C Test Spool Valve Failure, April 16, 1984
-- MR 84-565, SRV 'D' Position Indication, April 16, 1984
-- MR 84-599, Hydrogen-Oxygen Analyzer Trouble, April 23, 1984

No violations were identified. The following items required inspector followup review.

a. MR 84-469, RHR Valve V10-39A Failure

The control room operator closed the torus cooling upstream isolation valve, V10-39A (RHR 39A) at 5:50 A.M. on April 3, 1984 to secure torus cooling. About one minute after closing the valve, a 125% thermal overload condition on the valve was annunciated in the control room and the valve circuit breaker on MCC 9B tripped open. Subsequent licensee investigation determined that the closing torque switch had not stopped the motor when the valve reached the fully closed position. The motor remained energized and drove the valve into its seat. The valve motor circuit was found to have failed and shorted, which caused the magnetic overcurrent trip to actuate. The motor operator housing was found cracked open and dislocated sufficiently to cause the internal mechanism to jam and prevent engaging the manual operator to move the valve by hand. The high thermal overload switch was set at 300% of the full load current value and it did not actuate to de-energize the valve motor. RHR 39A is a 12 inch gate valve with a 2.6 HP motor and an SMB-0 operator.

The licensee inspected the valve stem to verify it had not been damaged. The valve body was inspected using an ultrasonic technique to assure the valve disc travelled freely with the stem. A replacement Peerless motor was installed along with a spare operator. Site engineering reviewed the difference between the old and new motor-operators to assure no performance characteristics would be changed. The valve was subsequently tested satisfactorily per OP 5520 and was stroke time tested.

The replacement motor was taken from service water valve V70-55B, which is in a non-safety class portion of the service water system. Onsite engineering reviewed the use of the replacement motor in a Safety class component and concluded that the quality of the replacement motor was no less than that of the original, since both were obtained from the same vendor under the same purchase order.

The inspector noted during a conversation with the Plant Manager on April 3, 1984 that a service request has been issued to the offsite engineering group to perform an evaluation of motor operated valves at the plant and to make recommendations for short and long term improvements to enhance reliability of safety related valves. This action was taken after licensee review of the operating history of motor operated valves. The results of this evaluation will be followed by the NRC to review the licensee's long term corrective actions (IFI 84-08-08).

No violations were identified.

b. MR 84-480, Serivce Water Valve V70-55B

This non-safety related maintenance request was issued on April 5, 1984 to remove the motor from service water valve V70-55B for use on RHR-39A. The valve is one of two series valves that control emergency make-up water to the condenser, as described in FSAR Section 10.6. The licensee declared the condenser fill line inoperable to facilitate maintenance on the RHR 39A valve, even though the 55B valve could still be operated manually. The licensee expected to have a replacement motor available for the 55B valve within about two weeks.

The inspector met with the Operations Superintendent on April 5, 1984 to discuss the status of the 55B valve. The inspector noted that removal of the 55B motor for longer than the specified two weeks would constitute a design change that would require processing as a Plant Alteration Request in accordance with AP 6000. The licensee noted the inspector's comments. A replacement motor was installed on the 55B valve and the valve was returned to an operable status on April 17, 1984.

No violations were identified.

c. MR 84-565, SRV 'D' Position Indication

Two of three pressure switches on the discharge piping of the 'D' safety relief valve (SRV) failed to reset after the valve lifted and closed on April 16, 1984. By design, any two of three switches must activate at 40 psi increasing to cause the valve 'open' lights to illuminate. The licensee found that both switches did not reset until 2 psi decreasing, which was far in excess of the allowable reset band. No qualified replacement switches were immediately available on site. The licensee exchanged switch 2-71-1C1 with switch 2-71-1D1 to provide for an operable position indication status for all SRVs. In the final configuration, all switches will operate properly to show open and closed position status.

No violations were identified.

d. MR 84-564, MSIV 80C Test Spool Valve

Licensee investigation of the MSIV 80C failure during the 10% closure test on April 16, 1984 determined that the test spool valve had failed in the ported position, which allowed for full closure of the main steam isolation valve. Upon disassembly of the spool valve, the licensee found oil and fine particles in the operating mechanism and the valve operated stiffly. The spool valve was replaced. The failed spool valve was cleaned, rebuilt and returned to stores as an operational spare.

No violations were identified.

8. Review of Plant Evolutions, Trips and Events

The inspector reviewed events that occurred during the inspection to verify continued safe operation of the reactor in accordance with the Technical

Specifications and regulatory requirements. The following items, as applicable, were considered during the inspector's review of plant trips and operational events:

-- observations of plant parameters and systems important to safety to confirm operation within approved operational limits;

-- description of event, including cause, systems involved, safety significance, facility status and status of engineered safety feature systems;

-- circumstances associated with the release of radioactive material and actions to control and contain the material;

-- verification of proper actions by plant personnel and verification of adherence to approved plant procedures; and,

-- verification that notifications were made to the NRC and in accordance with 10 CFR 50.72 and 50.73, as applicable.

Items reviewed during this period included a reactor scram and main steam line isolation at full power on April 16, 1984.

a. Reactor Scram on April 16, 1984

The main steam isolation valves automatically closed in response to a high steam flow condition that occurred with the reactor 100% full power (FP) on April 16, 1984. The high steam line flow condition occurred in steam lines A, B and D as a result of a full closure of the 80C MSIV during a 10% closure surveillance test. The control room operator tried to avert the Group 1 isolation and a scram by rapidly decreasing reactor recirculation flow using the master manual flow control mode. However, a reactor scram occurred at 7:42 A.M. with reactor power at about 87% FP. The surveillance test requires exercising each MSIV 10% shut and subsequently reopening the valve. After the shut signal was removed, the 'C' inboard MSIV continued to close until it was fully shut due to a faulty test spool valve.

Following the scram, reactor water level decreased to about 125 inches (referenced to the top of the active fuel), recovered to about 185 inches and then stabilized within the band of 140 to 160 inches. Reactor pressure increased to about 1080 psig, which caused the 'A', 'B' and 'D' safety relief valves to lift. The fourth valve was not required to open. All valves subsequently reseated in 10 seconds. However, the position indication for the 'D' SRV failed in the open position. The control room operator verified proper closure of the valve by noting the decreasing tailpipe temperature. The MSIV isolation was cleared at 7:44 A.M. to recover the main condenser as the reactor heat sink.

No other safeguard systems were called upon to operate. Plant systems other than those noted above responded as designed. The reactor was brought critical during the evening of April 16, 1984 after corrective maintenance on the failed components and completion of a post trip review in accordance with AP 0154. The inspector had no further comments on this item.

No violations were identified as a result of this review. See paragraph 5 above for further discussion of the licensee's post scram recovery actions.

9. Staffing and Organizational Changes

a. Staffing Changes

The licensee announced the following staffing changes during the inspection period:

+ Mr. J. Babbit was assigned to the position of onsite Security Supervisor.

+ Mr. W. Wittmer was appointed to the position of Project Manager for the recirculation pipe replacement program, a position he is expected to hold until completion of the program in 1986. The Plant Manager stated during a meeting on April 24, 1984 that the Department reporting lines would remain fixed in the temporary realignment established on February 2, 1984 with the Maintenance and Instrument and Control groups reporting to the Operations Superintendent.

  The inspector noted the potential conflicts with the organization structure described in the Technical Specifications should the 'temporary realignment' continue indefinitely. The Plant Manager stated that a decision will be made after the upcoming refueling outage to either resume the originally established reporting lines or to process a permanent organization change. Licensee staffing will be reviewed further during a subsequent routine inspection (IFI 84-08-09).

b. Fire Brigade Qualifications

The Shift Engineers (SE) are used as the Fire Brigade Leader at VY. The SE together with two Auxiliary Operators and two members of the guard force comprise a minimum 5 man fire brigade for each shift. The licensee recently instituted new administrative requirements for the Fire Brigade that requires members to pass a cardiovascular stress test to qualify for brigade duty. The results of stress tests completed during the week of April 30, 1984 disqualified 4 of 7 SEs and 3 auxiliary operators from fire brigade duty.

The loss of 4 SEs required that alternate assignments to be made and overtime be used to meet the minimum 5 man brigade requirements. The Operations Shift Schedule for weeks ending April 28, May 5 and May 12, 1984 was discussed with the Operations Administrative Assistant to review the use of overtime. Overtime used during this period was in accordance with the overtime limitations specified in AP 0036.

The licensee considered the present staffing problem to be a temporary one as measures are taken to re-qualify personnel for brigade duty. Fire brigade staffing will be reviewed further during subsequent routine inspections.

No violations were identified.

10. Confirmatory Order Dated June 27, 1983

By letter FVY 84-22 dated March 13, 1984, the licensee submitted his plans for inspection and modification of the recirculation and other reactor coolant pressure boundary piping during the refueling outage scheduled to begin on June 16, 1984. This action was in accordance with the NRC Confirmatory Order dated June 27, 1983, to submit such plans at least 3 months prior to the start of the outage. The licensee's submittal has been received by the NRC:NRR staff and is under review to determine the acceptability of the licensee's proposed inspection program and plans. This item will be followed on a subsequent inspection (IFI 84-08-10).

11. Unresolved Items

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 5 of this report.

12. 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.