U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Report No. 84-10

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: May 8 - June 4, 1984

Inspectors: William J. Raymond 6/25/84
W. J. Raymond, Senior Resident Inspector

Robert M. Gallo 6/28/84
R. M. Gallo, Chief, Reactor Projects
Section 2A, Projects Branch 2

Inspection Summary:
Inspection on May 8 - June 4, 1984 (Report No. 50-271/84-10)

Areas Inspected: Routine, unannounced inspection on day time and backshifts by the resident inspector of: actions on previous inspection findings; plant power operations, including operating activities and records; plant physical security; surveillance testing; maintenance activities; refueling outage preparations; and followup of recirculation system decontamination test results. The inspection involved 61 inspection hours onsite by the resident inspector.

Results: No violations were identified in 6 of the 7 areas inspected. One violation was identified in the area of surveillance testing controls, concerning the failure of Instrument and Control personnel to secure from testing in accordance with OP 4374 on May 8, 1984 (paragraph 8).

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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. J. Pelletier, Plant Manager
Mr. P. Donnelly, Instrument and Control Supervisor

2. Status of Previous Inspection Findings

a. (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 were any further depositions of contaminated material outside the plant radiation controlled area. All areas surveyed were found to be clean with measured radiation levels less than the general area background level of 1000 disintergrations per minute (dpm), except as noted below.

Minor spots of contamination were identified at fourteen locations. The contamination levels were in the range of 5,000 to 50,000 dpm as measured with an RM-14 survey instrument with an HP-210 probe. Three of the fourteen spots of contamination were located at the protected area fenceline. Two of the fourteen locations contained 'hot spots'. One hot spot reading 300,000 dpm (1.0 mR/hr) was found at the West protected area fence, about 15 feet North of the Vehicle Gate. A second hot spot reading 0.2 mR/hr was found adjacent to the North Warehouse at the site where the lump of contaminated material was first discovered on February 2, 1984. A spot with readings of 12,000 dpm was initially identified as contamination on the ground beneath a non-radioactive, spent oil storage tank in the Southeast sector of the site. Subsequent investigation determined that the measured radioactivity was coming from within the tank.

The source of the contamination found in three locations at the protected area fenceline is unknown. The contamination found at three locations in the Southeast sector of the site was most likely caused by contaminated material previously stored in the area. The contamination found at two locations under and near the AOG catwalk was most likely caused by personnel traffic out of the radiation controlled area at that location. The contamination found at five locations near the North Warehouse was most likely caused by the spill of material that was identified on February 2, 1984. The activity found in the spent oil storage tank appears to be evidence of a previous uncontrolled disposal of radioactive material.

The contaminated areas were cleaned up as they were found. The Chemistry and Health Physics Supervisor stated that the contents of the spent oil storage tank were removed for disposal as radwaste. The licensee's isotopic analyses of the contamination identified all material discovered during the survey to be reactor corrosion and activation products i.e., Co-60, Cs-137, Mn-54 and Zn-65.

A small area of the site access road between Gatehouse #1 and Route 142 was surveyed in the presence of Vermont State representatives on May 14, 1984. No readings above background were identified.

The licensee's review of the survey findings was in progress at the conclusion of this inspection. The licensee's evaluation of the survey results will be followed by the inspector during future routine inspections.

b. (Open) Follow Item 84-08-03: Evaluation of Degraded Station Battery Conditions. The licensee replaced Cell #11 in the 'A' Station Battery on June 1, 1984, as discussed in paragraphs 6 and 7 below. The decision to replace cell #11 was made after an engineering review concluded that some uncertainty existed as to how the battery bank would perform with the degraded cell under full load conditions. Each station battery is comprised of 60 LC-33 cells manufactured by the C&D Company.

The inspector met with the Maintenance Senior Engineer on May 17, 1984 to review the information then available on the battery problem. The degradation in voltage and specific gravity for cell #11 had been noted during surveillance testing in April, 1984. Although no degradation in cell parameters had been previously noted, anomalous conditions had been observed in the past and were subsequently monitored. The C&D Battery Company inspected the 'A' and 'B' station batteries in 1976 and noted a copper brown discoloration on the battery plates at that time. The C&D Company inspected the batteries again in 1978 and took one cell back to its facilities for examination. The vendor concluded that the observed discoloration was caused by copper contamination of the plates. The extent of the contamination at that time was deemed not to be a concern. The vendor's conclusions were documented in a December 26, 1978 letter to the licensee.

The observed discoloration on the lead coated negative battery plates is caused by copper contamination in the electrolyte. The vendor believes that the origin of the copper is from either the battery post insert or from copper sulfate particles that originate from intercell connections. The effect of the deposits on the negative plates is to degrade the cell voltage over time. As of May, 1984, degraded cell performance had occurred only on cell #11 of the 'A' station battery. There have been no adverse trends observed in station battery performance based on the operating cycle discharge tests. There have been no adverse trends observed on other battery cells. Preliminary inspections of both batteries in May, 1984 identified various stages of plate contamination in both batteries, as follows: station battery 'A' had 18 cells with copper discoloration and 2 cells with discoloration and copper precipitation; station battery 'B' had no cells with precipitation and 11 cells with discoloration. A more detailed battery inspection will be conducted during the refueling outage. Cell #36 of the 'A' battery will also be changed out during the outage because of the amount of precipitate observed in the cell.

The licensee was recently notified by the Exide Company of potential problems with the batteries associated with the 'A' and 'B' uninterruptible power supplies (UPS). These batteries were first installed in 1977 and then replaced in 1978 due to a problem with the hydrogen seal cap installed around the cell posts. An inspection by the battery vendor earlier this year also identified copper discoloration of the negative plates which suggests a copper contamination problem exists. However, the vendor has not provided a written report of his inspection to the licensee, or an engineering evaluation of the observed conditions and a recommendation for corrective action. Preliminary information from the vendor's service representatives indicates that the copper contamination may be due to an electrolyte interraction with the cell post insert. The licensee stated that the Exide Company has previously filed a Part 21 report for problems associated with its G-series of batteries. Vermont Yankee uses the E-series battery.

The inspector examined the UPS batteries with a licensee representative on May 17, 1984. Various stages of copper discoloration were noted on several cells in the 'A' and 'B' battery banks. The inspector also noted that several cells had oxide encrustations built up along the side of the cell post to the extent that the hydrogen seal caps were cocked and cracked. The licensee has noted this condition and will address the effects of increased stress on the cell posts.

The licensee's evaluation of the causes for the degraded conditions on both the UPS and main station batteries is still in progress, along with a determination of what additional corrective actions may be required. This item remains open pending completion of the above actions and subsequent review by the NRC.

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 physcial 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. The inspector also reviewed the status of security systems and the completion of compensatory measures for a security event on May 30, 1984.

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: (1) selected Technical Specification limits were met; (2) log entries involving abnormal conditions provided sufficient detail to communicate equipment status, correction, and restoration; (3) operating logs and surveillance sheets were properly completed and log book reviews were conducted by the staff; (4) Operating and Special Orders did not conflict with Technical Specification requirements; and, (5) Jumpers (Bypasses) did not create discrepancies with Technical Specification requirements and were properly approved prior to installation.

The following plant logs and operating records were reviewed periodically during the period of May 8 - June 4, 1984:

- Shift Supervisor's Log
- Night Order Book Entries
- Control Point Log
- Jumper/Lifted Lead Log
- Maintenance Request Log
- Switching Order Log
- Shift Turnover Checklists
- Radiochemistry Analysis Log
- RE Log Typer-Core Performance Log
- Chemistry Log
- Discharge Permit Log through 84-367
- PRO Reports 8/84 and 13/84

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, the Intake Structure 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.

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. The inspector reviewed the installation of new cabling between the cable vault, the control room and the switchgear room on May 31, 1984 to assure that fire stops were installed for the new electrical penetrations.

No violations were identified.

c. 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 84-336 were reviewed and no discrepancies were noted.

No violations were identified.

d. 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 April 30, 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.

e. Safeguard System Operability

Reviews of the Residual Heat Removal, Residual Heat Removal Service Water, High Pressure Coolant Injection, Service Water Alternate Cooling Tower, Standby Liquid Control 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 violations were identified.

f. 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 Permits (RWPs) were 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 violations were identified.

g. 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. 84-36 through 84-39. The inspector noted that Request No. 84-39 was issued to repair flow transmitter 13-58 on the reactor core isolation cooling system on May 15, 1984. This item is discussed further in paragraph 6 below.

No violations were identified.

h. 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 Appendix C of OP 2115, Primary Containment. 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 and flanges.

No violations were identified.

i. 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. Boron analysis results recorded for the Standby Liquid Control System on May 5, 1984, were reviewed.

(1) Elevated Offgas Release Rates

Offgas release rates and reactor coolant iodine levels increased following the rod pattern adjustment on May 20, 1984. The increased radiation levels were evidence that fuel failed following the control rod movement, but not necessarily as a result of the rod adjustments. There were no increases in the stack release rates following the fuel failure due to the retention of the noble gases by the charcoal adsorbers in the Advanced Offgas system.

Plant operators first noted the elevated offgas release rates at 7:40 A.M. on May 20, 1984 when radiation monitors for the 'A' Guard Bed increased to the hi-hi alarm point. The VY Duty Officer was contacted in accordance with the alarm response procedure. Health Physics personnel were requested to obtain a grab sample from the offgas system to confirm the monitor readings. Sample results reported to the control room indicated that the offgas release rates had increased by about 3 times the normal levels to 2200 uCi/sec. There were no increases measured downstream of the guard beds or at the steam jet air ejectors. No further actions were called for by facility procedures or were required in response to the condition.

Reactor coolant iodine levels increased slightly to about 7.0 X E-3 uCi/ml. Release rates and activity levels remained well below the Technical Specification limits. Stack release rates remained less than the minimum detectable value of about 100 uCi/sec. The licensee trended the increased release rates. Subsequent evaluations to locate the quadrant of the failed bundle were unsuccessful. Reactor Engineering personnel concluded that fuel thermal parameters remained well within the preconditioning limits during the May 20, 1984 rod pulls. The licensee estimated, based on the level of increased activity, that one fuel pin in one bundle had failed. This preliminary estimate will be confirmed by fuel sipping operations during the refueling outage.

This item is considered open and will be followed during the outage to determine the results of the licensee's findings from the sipping operations and to determine the licensee's conclusions regarding the fuel failure mechanism (IFI 84-10-01).

(2) Sample System Operability

The Plant Manager reported the following information to the inspector on May 18, 1984. During an engineering review of the Post Accident Sampling systems installed in 1983 per the requirements of NUREG 0737 Item II.B.3, the licensee identified a system design deficiency in the containment atmosphere sampling system that would preclude obtaining a sample representative of iodine and particulate activity in the containment atmosphere during post accident conditions. The problem in obtaining representative samples was attributed to a condenser and a filter in the sample lines of the sample system. The filter and condenser were not properly considered during the original design engineering reviews. The system can obtain representative samples of containment gaseous activity.

The licensee further reported that discrepancies were identified in stack monitoring instrumentation. The discrepancies were characterized as being a type that would not preclude obtaining a representative measurement of stack activity and release rates.

The licensee's engineering review is in progress to determine the necessary corrective actions. These items were reviewed during an NRC Team inspection of the post accident sampling systems during the week of May 21, 1984. NRC concerns regarding the correction of these deficiencies are being tracked as part of Inspection Report 50-271/84-11.

No violations were identified.

6. Operational Status Reviews

The 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.

a. Recirculation Weld Leakage Detection System

The recirculation weld leakage detection system remained in a partially operable status during the inspection period, with status information available from six of seven detectors. The system was energized daily to check the status of the detectors. No indications of recirculation system weld leakage was detected.

The inspector noted that unidentified leakage into the drywell sumps increased from zero to about 0.15 gallons per minute during the inspection period. The trend in drywell leakage will be monitored during subsequent routine inspections.

No violations were identified.

b. Diesel Generator Trip During Testing

Plant operators started the 'A' diesel generator for routine surveillance at 1:25 A.M. on May 8, 1984. The diesel tripped off line automatically on loss of field flashing at 7:00 A.M. when the control room operator adjusted the main switchyard voltage from 354 to 361 KV in response to a request from the offsite load dispatcher. After a review of the diesel protection circuitry, the operators concluded that the voltage adjustment on the 345 KV switchyard caused a change in volts-amps reactive (VARs) in station equipment, which was sensed as a momentary loss of field flashing for the diesel. The diesel was restarted at 7:15 A.M. and the surveillance test was satisfactorily completed.

No violations were identified.

c. Loss of Post-Accident Monitoring Instrumentation

While performing shift turnover panel checks at 4:00 P.M. on May 8, 1984, the control room operator noted that both channels of torus narrow range level indicated out of specification high at 1.5 feet. Level channels LT 16-19-46A and 46B provide monitoring of torus water volume over the range of 0 to 3 feet, which corresponds to a volume of 68,000 to 70,000 cubic feet. The maximum torus water volume allowed by the Technical Specifications is 70,000 cubic feet, or about 1.25 feet on the narrow range instruments.

A calibration of both level channels was completed at the end of the previous shift. The pre-calibration readings on the channels was 1.1 feet, which is the normal torus level. The level signal for recorder 23-73 on control room panel 9-3 is derived from level channel 46B. Recorder 23-73 indicated that the level instruments read 1.5 feet immediately upon return to service following calibration. The operators noted that there was no change in torus volume as indicated by the wide range instruments. The high pressure coolant injection suction valves were found in the test configuration with suction taken from the torus. The suction valves were returned to the normal standby alignment with suction taken from the condensate storage tank.

The shift supervisor declared both channels inoperable and initiated plans to commence a reactor shutdown within 6 hours in accordance with Technical Specfciation 3.2.6, unless the instruments were sooner made operable. No credit was taken for the torus wide range level channels since the instruments are not recognized by the technical specifications. I&C personnel were called back to the site to check the calibration on level instruments 46A and B. A report was made to the NRC Duty Officer at 4:55 P.M. in accordance with 10 CFR 50.72 b (1) (v), loss of accident assessment capability. I&C personnel refilled the reference legs for both transmitters and re-verified the channel calibrations. Both narrow range level instruments were returned to an operable status by 5:50 P.M., with indications of 1.07 and 1.08 feet.

The licensee's subsequent review of this event determined that the level indications on both channels was satisfactory upon completion of the calibrations at about 3:20 P.M. on May 8, 1984. However, the level instrument reference legs are filled from a demineralized water supply header that passes through the 213 ft. elevation of the RCIC room. The 213 ft. RCIC room ambient temperature is normally elevated due to the presence of the turbine steam supply line. The licensee postulated that a hot slug of water was used to fill the reference legs. The indicated level changed as the reference leg water temperature cooled to the ambient temperature of the torus room. The licensee subsequently determined that this event was reportable to the NRC under 10 CFR 50.73.

The inspector has no further comment on this event in regard to plant operations. This item is discussed further below in the area of surveillance test controls.

d. <u>RCIC Flow Transmitter Calibration</u>

Plant operators declared the reactor core isolation cooling (RCIC) system inoperable at 9:55 A.M. on May 15, 1984 after I&C personnel determined that the discharge flow transmitter FT 13-58 could not be calibrated to within the tolerances prescribed by OP 5315. Alternate system testing was begun in accordance with the technical specifications. Testing of the high pressure coolant injection system was completed at 1:10 P.M. The flow transmitter was replaced at 1:35 P.M. and the RCIC system was returned to an operable status following testing at 6:05 P.M.

The licensee determined that the transmitter calibration was in error but conservative, such that slightly more than the nominal 400 gallons per minute flow would have been provided by the flow controller during automatic operation of the RCIC system. Thus, the RCIC system could have performed its intended function and was in fact operable with the faulty flow transmitter.

No violations were identified.

e. <u>Elevated River Levels</u>

Excessive rains in the area during the period from May 28-31, 1984 resulted in areas of local flooding and elevated levels in the Connecticut River. Local civil defense agencies reported that the river was 6 feet above flood stage for Southeastern Vermont on May 31, 1984. The Vernon Pond reached the 223 foot elevation as measured by markings on the side of the Intake Structure. Grade elevation for the plant buildings is 252 feet. The lowest Reactor Building elevation is 213 feet.

The adverse river and weather conditions created had no adverse impact on the plant or on plant operations. Plant operators were cognizant of and trended the river conditions. No action levels were reached in abnormal operating or emergency procedures. The first action level is reached in OP 3500 and AP 3021 when river level rises to the 235 foot elevation.

No violations were identified.

f. Inoperable Station Battery

Cell #11 of the 'A' 125 VDC station battery system was found degraded during surveillance testing on April 9, 1984 with measured voltage and specific gravity below the acceptable values of 2.13 and 1.19, respectively. The battery remained operable since the technical specifications allow one of the 60 cells to be out of service. Plans were made to replace the faulty cell during the refueling outage and a replacement was obtained from C&D Battery Company.

After further consultation with engineering and the battery vendor, the licensee concluded that it would be prudent to change out cell #11 prior to the outage. This decision was based on the uncertainties regarding the potential impact that the faulty cell could have on the rest of the bank.

Battery charger CA-1 is rated for 150 amps and carries the normal loads connected to the associated battery bus, DC-1. These loads include valve power for the high pressure coolant injection system, and instrument and control power for core spray train B, low pressure coolant injection train B, diesel generator B, 4KV switchgear 1 and 3, and 480V switchgear #8. However, the charger is load shed from DC-1 during a loss of normal power (LNP) condition, since the DC-1 loads for the combined LOCA-LNP condition can exceed 1600 amps. Thus, power for the systems fed from DC-1 would not be operable for the LOCA-LNP condition.

The licensee questioned the inspector regarding the equipment operability requirements specified in Technical Specifications 3.5 and 3.10 concerning operation with one station battery. After consultation with Region I management, the inspector informed the Plant Manager at 11:00 A.M. on June 1, 1984 that, with one station battery inoperable, the equipment supplied from or supported by that battery would also be considered inoperable.

The licensee declared the 'A' station battery inoperable at 2:40 P.M. on June 1, 1984 and removed the battery from service to replace cell #11.

Normal DC-1 loads were fed by charger CA-1. The battery was declared inoperable in accordance with Technical Specification 3.10.B.2, which allows for continued reactor operation for 3 days. However, portions of the core spray, low pressure coolant injection and high pressure coolant injection systems were also declared inoperable in accordance with Technical Specification 3.5, based on the NRC position presented above. This combination of equipment outages was in excess of that allowed by the technical specifications, which required that the plant be placed in cold shutdown within 24 hours. Additionally, the 'A' train of the standby gas treatment system was inoperable at the time pending completion of a satisfactory charcoal adsorption test following replacement of a charcoal cell on May 31, 1984.

A controlled power reduction was commenced. The reactor was operating at 96% power at the time in an end-of-cycle coastdown. An Unusual Event was declared at 2:45 P.M. in accordance with the emergency plan due to a plant shutdown required by Technical Specification limiting conditions for operation. Notifications were made to the offsite State agencies at 2:48 P.M. and to the NRC Duty Officer at 3:05 P.M.

Preparations to replace the cell were made and a replacement battery was in the battery room when the battery was declared inoperable at 2:40 P.M. The battery was immediately tagged out, disconnected from bus DC-1 and turned over to maintenance personnel. Maintenance personnel replaced cell #11 by 4:30 P.M. Voltage and specific gravity measurements on the replaced cell were taken and found acceptable at 2.07 volts and 1.214 specific gravity. The 'A' station battery bank was found acceptable and declared operable. The 'A' battery bank was reconnected to DC-1 at 4:45 P.M. The reactor power reduction was stopped at about 84% full power and the Unusual Event was terminated at 4:57 P.M.

The 'A' standby gas treatment train was subsequently tested satisfactorily and returned to an operable status at 11:15 P.M. on June 2, 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 and equipment return to service was proper, including the completion of operability testing.

- -- MR 84-451, Replacement of Cell #11 on the 'A' Station Battery
- -- MR 84-721, RCIC Flow Transmitter FT 13-58
- -- MR 84-632, HPCI Pressure Indicator PI 23-99
- -- MR 84-716, HPCI Pump Seal Leakage
- -- MR 84-643, Reactor Level Recorder 6-98
- -- MR 84-660, Chiller SCH-1-1 Refrigerant Leak
- -- MR 84-672, Reactor Level Transmitter LT 72 B&C Indication Error
- -- MR 84-717, RHR Flow Recorder 10-143 Inoperable

The following items warranted followup by the inspector.

a. Replacement of Station Battery 'A' Cell #11

MR 84-451 was issued on March 30, 1984 based on test results obtained per VYOPF 4210.01, which showed a gradual deterioration in the voltage and specific gravity readings for cell #11. The cell parameters were trended by maintenance personnel since a degraded specific gravity reading was noted during routine surveillance of the battery on April 2, 1984.

The inspector witnessed the replacement of cell #11 on June 1, 1984 in accordance with MR 84-451. The inspector also witnessed the subsequent readings taken on the cell and the 'A' battery bank. The final voltage and specific gravity readings were 2.07 and 1.214, respectively. The 'A' Station Battery bank voltage was found to be acceptable at 130.5 volts. The 2.07 volt reading on cell #11 was considered acceptable since the reading would increase about 2.13 volts after subsequent operation on the battery charger. The inspector reviewed the steps to replace the cell for conformance with the recommendations in the vendor manual, C&D Station Battery Installation and Operating Instructions. Rack and battery post bolts were torqued to the values recommended by the battery vendor using calibrated equipment.

No violations were identified.

8. Surveillance Activities

Surveillance testing completed on May 8, 1984 in accordance with OP 4374, Revision 13, HPCI-Torus Water Level Functional Test, was reviewed to verify that: testing was performed by qualified personnel; test data demonstrated conformance with Technical Specification requirements; the test procedure and methodology were adequate based on a review of facility flow and circuit wiring diagrams; and, system restoration to service was proper. The following items warranted further followup.

a. Torus Level Setpoints

Procedure OP 4374 provides for a calibration of the torus narrow range level instruments over the range of 0 to 3 feet, and further, provides for a verification that the high pressure coolant injection suction valves automatically switch to the torus when torus level reaches 1.92 feet.

The inspector noted that the 1.92 foot setpoint appeared to be in disagreement with Technical Specification Table 3.2.1, which indiates that the torus high water level trip setting must be less than or equal to 2 inches. The matter was discussed with the I&C Supervisor, who stated that the setpoints in both OP 4374 and the technical specifications are correct.

The setpoint difference was addressed in a memorandum from D. Phillips to R. Pagodin dated December 27, 1984. The setpoint value in the technical specifications refer to the location of the original float type level switches that were used to provide the transfer function. The level switches were subsequently replaced by analog instruments which use a differential pressure transmitter with an electronic trip unit. However, the technical specification values were never changed to reflect the different zero reference for the new instruments. The location of the original level switches was elevation 228' 2", which corresponds to an indication of 1.92 feet on the narrow range indicator. The technical specification value of 2" corresponds to an indication of 2.08 feet. The inspector further noted, based on the December 31, 1983 memo, that a similar situation exists for the condensate storage tank (CST) level instruments.

The insepctor noted that the above setpoint correlations was recently provided to the plant operators to clarify the apparent discrepancies between the installed instrumentation and the technical specifications. However, if left unchanged, this apparent discrepancy can be a future cause of confusion for plant operators when determining the appropriate CST and torus levels at which RCIC and HPCI suction valves should transfer. The technical specification setpoints should be revised to reflect the zero reference for the existing narrow range instruments. This item is open pending revision of the technical specifications to reflect the installed instrumentation for torus and CST level (IFI 84-10-02).

b. Restoration of Equipment Following Testing

During shift turnover checks at 4:00 P.M. on May 8, 1984, the control operators noted that both channels of narrow range torus water level were reading upscale. Additionally, the HPCI suction valves were realigned from the torus to the CST to provide for the normal standby line-up. The level instruments were declared inoperable per Technical Specification Table 3.2.6 and a six hour LCO was entered to shutdown the plant unless the instruments were sooner made operable. I&C personnel were called in to the plant to re-calibrate the instruments. Both narrow range channels were re-calibrated and returned to service at 5:30 P.M. and were declared operable.

The instruments channels were calibrated during the previous shift. Neither the Day or Swing shift supervisors were notified that testing activities had been terminated. The following information was obtained based on discussions with the I&C Department Supervisor on May 9, 1984.

Calibration of the torus level instruments started at 2:45 P.M. on May 8, 1984 in accordance with OP 4374 and test activities up to Step 11 were completed at 3:25 P.M. The Operations crews completed a shift turnover during the intervening period. The narrow range level instruments were returned to an online status when test activities were terminated at Step 11. The post-calibration torus level indication was verified to be satisfactory by test personnel. Test personnel then left the control room in accordance with the normal practice to observe a two hour wait period prior to final sign-off of Step 11 that the post-test level indication was satisfactory. The wait period was observed to account for thermal stabilization of the reference legs after refilling the legs during the test. The narrow range level instrument readings were checked again by test personnel at 3:30 and 3:45 P.M. and were found acceptable. Test personnel went home for the night with the intention of finishing Steps 11, 12 and 13 of OP 4374 the next day.

Procedure Steps 11 through 13 include a final verification that the level instruments have been properly returned to service, that annunciators and relays energized during the test have returned to the normal status, and that the HPCI suction valves are realigned per the discretion of the shift supervisor. The final step of OP 4374 requires that the Shift Supervisor review and sign for the completed test results. The Shift Supervisor review of the test result, along with any noted discrepancies, is intended to verify that equipment and components important to safe plant operation were found to be operable by the test. Test personnel apparently verified that the final status of the annunciators and relays was acceptable, even though these steps were not signed off in the procedure. The control room operator was notified that testing was done at 3:45 P.M., but that final sign-off of the test results would wait until the stabilization period was completed. The shift supervisor was not notified since he was not in the control room at the time.

The termination of test activities at 3:45 P.M. on May 8, 1984 without formal notification of the Department or Shift Supervisors constituted a failure to complete testing requirements in accordance with OP 4374. The termination of test activities circumvented the administrative controls established to assure that the Shift Supervisor was aware of the completion of test activities in progress on his shift, the test results and the operability status of safety related equipment. The failure to complete testing in accordance with OP 4374 is contrary to the requirements of Technical Specification 6.5 (VIO 84-10-03).

9. Preparations for Refueling

The licensee issued a draft copy of the 1984 Outage Manual on May 15, 1984. The inspector reviewed the manual to determine the scope of work activity planned for the outage, along with the licensee's organization and directives established to control the activities. No inadequacies were identified. The work schedules presented in the manual will be used to track the progress of the outage and to schedule support inspection of outage activities.

No violations were identified.

10. CERT Results

The Plant Manager informed the inspector on May 16, 1984 of preliminary information that was reported to the licensee by London Nuclear regarding the constant expansion rate testing (CERT) recently completed by Ontario Nuclear to qualify continued use of the Can-Decon operation to decontaminate the recirculation system. The test results were considered to be preliminary pending further review of the results by the licensee's contractors.

The licensee contracted London Nuclear to study the effects of prolonged exposure to the Can-decon process to support continued use of the decontamination method during the upcoming refueling outage. The process was used to decontaminate the recirculation system during the 1983 refueling outage. Prior to the 1983 decon operation, CERT testing was completed following a 250 hour exposure of test materials to the decontamination solution. The results of 250 hour test showed no increased sensitivity of the test material to intergrannular stress corrosion cracking. The Can-decon solution was used in the recirculation system during the 1983 refueling outage. The exposure time of recirculation system components to the decon solution was within the 250 hours qualified by the 1983 test. The limiting exposure occurred to the recirculation pumps at 250 hours.

Test coupons were exposed to the Can-decon solution for 500 hours during the most recent tests and then tensile stressed. The results from the constant expansion rate test indicated the coupons exposed for 500 hours exhibited intergranular stress corrosion cracking more quickly than a control group of samples that were not exposed to the solution. The inspector noted that the acceptability of the 1983 decontamination has not been changed by the 500 hour test results.

The licensee stated that the 500 hour CERT results were considered preliminary pending receipt of the formal written report from the contractor, along with the contractors evaluation of the test data. The test coupons used in the tests are selected to produce a bounding set of test results that are representative of a large population of recirculation system piping and components. The preliminary results are therefore not necessarily representative of how actual plant piping would respond to the solutions, based on the conservative nature of the initial tests. The licensee stated that more information on the tests is expected from his contractor and that this information would be made available for inspection review.

The inspector stated that this item is open pending completion of the licensee's review and evaluation of the 500 hour Can-decon test results, and subsequent review by the NRC (IFI 84-10-04).

11. Management Meetings

Licensee management was periodically notified of the preliminary findings by the resident inspector during the inspection period. A summary was also provided at the conclusion of the inspection and prior to report issuance.