U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Report No. 84-12

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: June 5 - July 30, 1984

Inspectors: William J Raymond 9/11/84
W. J. Raymond, Senior Resident Inspector

William J Raymond 9/11/84
for J. E. Beall, Project Engineer

William J Raymond 9/11/84
for J. R. Johnson, Resident Inspector

Lowell E. Tripp 9/20/84
Lowell E. Tripp, Chief, Reactor Projects
Section 3A, Projects Branch 3

Inspection Summary:

Inspection on June 5 - July 30, 1984 (Inspection Report No. 50-271/84-12)

Areas Inspected: Routine, unannounced inspection on day time and backshifts by resident and Region-based inspectors of: action on previous inspection findings; physical security; routine power operations and refueling outage activities; maintenance activities; surveillance testing; response to events; refueling activities; inservice inspections and recirculation system examinations; plans for the 1984 emergency preparedness drill; and, followup of worker concerns regarding contractor medical screening. The inspection involved 229 hours on site by resident and regional inspectors.

Results: No violations were identified.

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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. D. Reid, Operations Superintendent
Mr. J. Pelletier, Plant Manager

2. Status of Previous Inspection Findings

a. (Closed) Follow Item 83-17-02: Review Long Term Corrective Actions For Valves V70-22A through 22D. The licensee approved Plant Design Change Request (PDCR) 84-01 which includes two major portions: 1) overhaul and refurbishment of valve internals with an epoxy corrosion inhibitor and 2) the addition of a three-inch bypass line around the original six-inch heat exchanger outlet valves. This modification will provide for better throttling capability for the cooling flow and reduce the previously indentified vibration problems. This PDCR is being tracked by the licensee for completion during the present refueling outage. This item is closed.

b. (Open) Follow Item 84-08-03: Evaluation of Degraded UPS Batteries. The licensee submitted a report under 10 CFR Part 21 on June 30, 1984 based on information received from the Exide Corporation which indicated that a defect may exist with respect to the Uninterruptable Power Supply (UPS) Batteries. The UPS batteries are used to power low pressure coolant injection valves during accident conditions. Vermont Yankee uses two redundant UPS batteries, each consisting of 192 'E' size cells, type EC11, that were purchased in 1979.

The battery vendor performed a detailed field inspection of the UPS batteries on March 28, 1984. The inspection revealed cracked terminal seals on many cells, which resulted from acid corrosion of the lead posts. The corrosion results in the formation of lead oxide encrustations which enlarge with time. The corrosion can cause additional stress on the plastic components of the cover and may expose the copper insert inside the battery post. Chemical interactions between the electrolyte and the copper insert causes the copper to go into solution and can result in copper contamination of the negative battery plates. The contamination indicates that the current carrying capacity of the battery post is degrading and that the cells are approaching an advanced end-of-life condition.

Exide reported the results of their inspections to the licensee by letter dated May 14, 1984. Exide determined that the present batteries could still perform their safety function, but strongly recommended that the batteries be replaced during the 1984 refueling outage. This recommendation was based on the vendor's evaluation which concluded that the UPS batteries were deteriorating and the ability to satisfactorily accomplish the safety function up to the 1984 outage could not be guaranteed.

After further review of the information received from Exide, the licensee concluded that the conditions observed at Vermont Yankee may be applicable to other users of the Exide 'E' Series batteries. The licensee notified the Resident Inspector on June 22, 1984 of his intent to submit a Part 21 report. The licensee has purchased two replacement 192 cell UPS batteries that use a new seal design. Licensee actions to replace the cells during the current outage are discussed further in paragraph 9 below.

The licensee has yet to determine the cause for the accelerated deterioration observed on the old UPS batteries, and to identify what changes have been incorporated in the new 'E' series design to correct this problem. This item will remain open pending completion of the licensee's evaluation regarding the UPS batteries and subsequent review by the NRC.

c. (Closed) Follow Item 83-26-05: Operability of Valve SW-16B. The licensee revised OP 4181, Service Water Valve Operability, Revision 9, to include manual valve SW-16B in the list of valves tested. SW-16B was opened manually in less than 10 minutes during a test per VYOPF 4181.02 on July 4, 1984. This item is closed.

d. (Open) Violation 84-05-02: RCIC 20 Logic Testing. The licensee responded to this item by letter FVY 84-56 dated June 1, 1984 and took exception to the alleged violation. The response was discussed with the Operations Superintendent during a meeting on June 13, 1984. The licensee stated in his response that previous NRC reviews of the controls in AP 0155 (as documented in Inspection Reports 77-09, 80-19 and 81-19) identified no concerns and provided apparent concurrence that the established philosophy was acceptable. The inspector stated that previous NRC inspections did not endorse "exceptions" to approved procedures that would change the "intent" of the procedures, or otherwise endorse "on-the-spot" changes made with less than the level of review specified by the technical specifications. The inspector stated that the licensee's response did not fully address the issues discussed in Inspection Report 84-05 and that this item would be the subject of further discussion in separate correspondence with the licensee.

The inspector also noted that the licensee's response did not address testing requirements for the RCIC 20 valve. The Operations Superintendent stated on June 13, 1984 that the testing issue was excluded by oversight and that a requirement will be added to OP 4100 to fully test the RCIC 20 logic. The inspector witnessed the performance of OP 4100, ECCS Integrated Automatic Initiation Test, Revision 9, on August 1, 1984 and noted that proper operation of the logic was demonstrated when the valve automatically opened in response to a low reactor vessel water level signal. Testing of the RCIC 20 logic is considered a resolved issue.

Resolution of inspection item 84-05-02 remains open pending further review by NRC management and subsequent discussion with the licensee.

e. (Open) Follow Item 84-10-02: Torus Level Instrument Setpoints. The discussion of this item on page 15 of the report contained an error in the reference to a memorandum that addressed the issue. The correct reference should have referred to a memorandum dated December 27, 1983 to the Engineering Support Supervisor, that addressed the different torus level setpoints used in the technical specifications and in plant procedures.

This item remains open pending completion of the licensee actions addressed in Inspection Report 84-10.

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. The inspector also reviewed the status of security systems and the completion of compensatory measures for a security events on June 11, 12 and July 3, 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 June 5 - July 30, 1984:

- Shift Supervisor's Log
- Night Order Book Entries
- Control Room Operator Log
- Auxiliary Operator Log
- Control Point Log
- Jumper/Lifted Lead Log
- Maintenance Request Log
- Switching Order Log
- Shift Turnover Log
- Radiochemistry Analysis Log
- RE Log Typer-Core Performance Log

No violations were identified.

5. Inspection Tours

The plant was operating in end of cycle coastdown at 94% full power on June 5, 1984 and a controlled shutdown began at 4:15 P.M. on June 15, 1984 to begin the 1984 refueling and maintenance outage. The plant entered the cold shutdown condition at 6:05 A.M. on June 16, 1984.

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, Drywell, the Intake Structure and the grounds within the Protected Area. Control room staffing was reviewed for conformance with the requirements of the Techncial 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. Work controls established under the following fire permits were reviewed to verify conformance with requirements of the permits: 84-333, 322, 432 and 354. No inadequacies were identified, except as noted below.

The personnel actions contributing to and following the activation of the turbine loading bay deluge system at 10:15 A.M. on June 12, 1984 were reviewed. The deluge system activated due to the operation of an electric drill in the bay. No inadequacies were identified.

A plant worker notified the inspector by telephone at 7:30 P.M. on July 20, 1984, of his concerns regarding working and general safety conditions in the drywell. No radiological concerns were identified. The drywell working conditions were reviewed by the inspector during an inspection tour on July 22, 1984. Housekeeping, general working conditions and radiological controls were acceptable.

Controls established per fire permit 84-432 for welding operations were reviewed during a tour of the drywell on July 22, 1984. A fire watch was established as required by the permit, but the person assigned to the duty at 12:00 P.M. did not have a copy of the permit and was therefore not familiar with its requirements. The individual was first assigned fire watch duty for the drywell on July 22, 1984. This matter was discussed with the Shift Engineer and Construction Superintendent for the shift. A copy of permit 84-432 was later found posted at the entrance to the drywell and reportedly had been posted there since its issuance on July 10, 1984. The activities of the fire watch met the requirements of the permit for the work in progress at the time, except that tours of the area were conducted at 5 instead of 15 minute intervals.

The inspector's concerns regarding fire watch duty and ALARA control were discussed during a meeting with the Operations Superintendent on July 30, 1984.

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 84-564, 577, 583, 627 and 628 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 June 15, 1984, there were (1) no deviations in excess of 0.10 from the established steady state normalized value; and (2) no failures in the 16 thermocouples initially installed on the 4 feedwater nozzles. The plant was shutdown on June 15, 1984 and no data was collected from the feedwater nozzles until the plant returned to full power steady state operations.

The licensee completed non-destructive examinations of the four feedwater nozzles during the outage in accordance with his commitments to NRR. The results of the examinations reported by the licensee on June 26, 1984 identified no flaw indications on any of the nozzles. This item is discussed further in paragraph 14 below.

No violations were identified.

e. Safeguard System Operability

Reviews of the Residual Heat Removal, Residual Heat Removal Service Water, High Pressure Coolant Injection, Core Spray, 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: (1) verification that each accessible, major flow path valve was correctly positioned; (2) verification that power supplies and electrical breakers were properly aligned for active components; and, (3) 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. Radiation work permits (RWPs) were reviewed to verify conformance with procedure AP 0502. Work activities in progress were reviewed for conformance with the RWP requirements. Radiation surveys were conducted by the inspector during plant tours to confirm proper posting of radiological areas. Work activities associated with the following RWPs were reviewed: 658, 701, 730, 1727, 1748, 2050-2053, 2056, 2057, 2060, 2062, 805, 806, 830, 1350 and 1333. No inadequacies were identified.

During fuel handling and sipping operations on June 22, 1984, the duty HP technician observed bubbles rising to the surface of the spent fuel pool. The results of the routine air sample taken over the pool at 4:00 A.M. showed a gas activity of about 2.4 E-5 uCi/cc from two unidentified isotopes. All personnel left the refueling floor. A second grab sample showed no elevated activity levels. Refueling floor general area air samples taken at 5:35 A.M. and periodically thereafter showed lower airborne concentrations at 7.0 E-10 uCi/cc. The calculated immersion dose rate corresponding to the 4:00 A.M. sample concentration was less than 20 mRem/hr. The actual dose received by personnel on the floor was much less than 20 mRem due to the short stay time. No inadequacies were identified.

During the removal of tools from the refueling floor at 7:10 A.M. on July 3, 1984, portions of the floor on the 303 ft. and 318 ft. elevations became contaminated. Levels up to 10,000 counts per minute were measured based on a "sweep" of the area with masslinn. The floor under the stairway to the 345 ft. elevation was also contaminated to levels up to 300,000 dpm/100 sq-cm. The inspector reviewed the actions taken by the licensee to post and control entry to the contaminated areas until the contamination was removed. No inadequacies were identified.

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. 84-40 through 84-50, 84-52, and 84-54 through 84-61.

Implementation of mechanical bypass request 84-14 on July 17, 1984 was also reviewed during this inspection period. The results of that review are documented in NRC Region I Inspection Report 50-271/84-20.

No violations were identified.

h. 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.

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 adjustments. The licensee later determined that a single fuel pin in one fuel bundle had failed, as discussed further in paragraph 11 below. There was no increase 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 (AOG) system. Offgas release rates remained stable at about 1300 uCi/sec through the end of the operating cycle.

Stack gas release rates did increase following the plant shutdown after the AOG system was taken off line. The stack gas release rates for the period from June 15-19, 1984 were reviewed and found to be much less than the value allowed by the technical specifications for the isotopic mixtures in the discharge. The maximum release rate occurred over a 45 minute period starting at 5:30 A.M. on June 16, 1984. The maximum release rate was 1.1 E+4 uCi/sec, which was much less than the technical specification limit of 1.0 E+6 uCi/sec for the mix of noble gases with an average disintergration energy of 0.08 Mev/disintegration.

No violations were identified.

6. Operational Status Reviews

The operational status of standby emergency systems and equipment aligned to support routine plant operation was confirmed by direct review of control room instrumentation. Control room panels and operating logs were reviewed for indications of operational problems. 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.

Unidentified leakage into the drywell sumps remained at about 0.15 gallons per minute until the plant was shutdown on June 15, 1984. An inspection of drywell conditions by the licensee following the plant shutdown identified the containment spray header as the leakage source.

No violations were identified.

b. Stack Gas Detector Operability

The Operations Superintendent informed the inspector on June 18, 1984 that the detector for the Stack Gas II instrument channel was found to have an aluminum "window", instead of a mylar one. The aluminum window would affect the detector response to beta emitters and thus cause an inaccurate response to noble gas releases. Additionally, the correlation used to convert detector response to a stack release rate was invalidated. The detector was changed in March 1984 by I&C personnel during routine preventive maintenance on the detector. Based on the above, the Stack Gas II instrument channel is considered to have been inoperable since March 1984. A new detector with a mylar window was installed and the channel was recalibrated.

The licensee also determined on June 17, 1984, that the Stack Gas I channel was indicating low by about 1/2 a decade when compared to grab sample results of stack effluents. The Stack Gas I problems were subsequently traced to a faulty circuit board in the detector control circuitry. The circuit card was replaced on June 23, 1984 and the detector channel was recalibrated.

Technical Specification 3.9.B.1 permits indefinite plant operations with one of two stack gas instrument channels inoperable. With both stack gas instruments inoperable, continued plant operations is allowed for up to 7 days provided supplemental instruments are operable to verify applicable release rates are met. The licensee reported the above detector problems to the NRC in licensee event report (LER) 84-10 dated July 16, 1984 in accordance with 10 CFR 50.73(a)(2)(i) as a violation of Technical Specification 3.9.B.1. The licensee stated in the LER that no release limits were exceeded.

This item is unresolved pending further NRC review of the stack gas instrument problems and the corrective actions taken by the licensee (UNR 84-12-01).

7. Plant Design Changes

The inspector reviewed the licensee's action involving Plant Design Change Requests (PDCR) and Engineering Design Change Requests (EDCR) in order to assure that the actions were in accordance with 10 CFR 50.59, the facility Technical Specifications, and the licensee's procedures.

The inspector reviewed records which included (1) the licensee's bases for concluding that the modification did not constitute an unreviewed safety question, (2) review and approval of the design change and Installation - Testing procedures, and (3) completed portions of installation procedures. The inspector also held discussions with the licensee's project engineers and implementing personnel, and toured the work sites in the reactor building.

Items reviewed included the following:

- -- The 1984 Refueling Outage Plan Manual
- -- PDCR 83-06, HPCI Automatic Suction Transfer Removal
- -- PDCR 83-07, Safety Valve Acoustic Accelerometer

-- PDCR 84-01, Isolation/Control Valves for Service Water to MGLO Coolers
-- EDCR 83-32, Fuel Pool Cooling EQ Modification, and
-- EDCR 84-413, Replace ASCO Solenoid Valves

The inspector determined that, in general, the licensee's program for implementing plant modifications was acceptable. Several comments and items requiring further review are described below.

a. During a review of the Refueling Outage Manual, the inspector noted several references to plant equipment that was found not to be qualified for the post accident environment. Examples were the RHR Service Water pump motor bearing cooling line solenoid valves, and the 24 VDC ECCS battery changers. Following discussion with the inspector, the licensee clarified these references and stated that this equipment was not known or found to be unqualified but that, because of the lack of ability to demonstrate environmental qualification, the equipment would be modified. The inspector acknowledged the licensee's statements and had no further questions concerning this area.

b. PDCR 83-06 - The safety evaluation for PDCR 83-06 dated January 14, 1984 stated that..."The HPCI system is safety class, but the automatic transfer of HPCI suction based on suppression pool level is non-nuclear safety related". FSAR Sections 6.2 and 6.4.1 state the following:

(1) Safety design basis 6.2.11 establishes the torus as the primary source of water for the core standby cooling systems (CSCS), so as to provide a closed (and thus, endless) cooling water path for core cooling during CSCS operation.

(2) Section 6.4.1 states (page 6.4-6) that the HPCIS initially injects water from the CST and then transfers its suction path to the suppression pool on either low CST level or high torus level to establish a closed loop for re-circulating water escaping from the break. The auto suction transfer arrangement satisfies safety design basis 6.2.11.

Based on the above, it appeared that the auto transfer of HPCI suction from the CST to the suppression chamber, on low CST <u>or</u> high torus level was a function needed to satisfy safety design basis 6.2.11, and was therefore safety related. The inspector discussed this item with the Senior Electrical Engineer on July 5, 1984. The licensee stated that the item would be reviewed and incorporated in the safety evaluation to determine whether the conclusions remained unchanged.

This item was addressed in Supplement 1 to PDCR 83-06 dated July 30, 1984. The licensee's revised safety evaluation concluded that the CST was the primary source of water for the HPCI system and that safety design basis 6.2.11 was satisfied on the basis of transfer on low CST level alone. The auto suction transfer on high torus level was found to be non-safety related. The inspector had no further comments regarding conformance of the PDCR safety evaluation with the assumptions presented in the FSAR.

Implementation of the PDCR 83-06 changes is contingent upon NRC staff approval of the licensee's proposed change to Technical Specification Tables 3.2.1 and 4.2.

This item is recognized in the PDCR package and is being tracked by the licensee. The inspector had no further comment on PDCR 83-06.

c. PDCR 83-07 - During a review of the installation and test procedure, the inspector noted that a system functional test was not included following the completion of the acoustic monitor modifications. The licensee's Senior Electrical Engineer stated that the routine surveillance functional test would be performed following the completion of this modification. Also, the inspector noted that seismic design calculations and drawings were included in the job order file and coupled with a paper clip to a cover sheet identified as "Attachment C" to the PDCR. The inspector stated that although this may meet the requirements of AP 6000, Section 4, for identifying "all" attachments as being part of the PDCR, there was a good possibility that some design documents could be inadvertently lost from the records package. The licensee acknowledged the inspector's concern and stated that this practice had not been a problem in the past because there is only one controlled PDCR package, and that an audit is performed by two project engineers prior to the closeout of the modification. The inspector had no further questions at this time.

d. PDCR 84-01 - The inspector noted that although the onsite review committee (PORC), the Plant Manager, and the Operational QA group had reviewed and approved the Installation and Test Procedure for valve and piping changes to the service water system, the hydrostatic testing pressure specified in Section 7.4 for the piping spool pieces was not in agreement with the test pressure specified in the design input document (PDCR Section 8.5.4). The hydrostatic test had not been performed yet and the licensee's cognizant engineer stated that this discrepancy would be corrected prior to the test. The inspector had no further questions.

e. EDCR 83-32 - This design includes changing the electric power supplies, level instrumentation and controls for the fuel pool cooling system to a safety classification and will assure that the fuel pool will not boil and contribute to the harsh environment following a LOCA. During the review process, several questions were raised by the licensee regarding system classification and seismic qualifications. A memorandum (VYB 84-105 dated April 23, 1984) was sent from the licensee to the Yankee Atomic Project Manager requesting resolution to concerns regarding correcting the safety classification of the fuel pool cooling system and whether the fuel pool cooling system should be qualified to withstand seismic events. The inspector will follow the licensee's resolution of these concerns which are expected by mid-July, 1984 (IFI 84-12-02).

f. EDCR 84-413 - This change includes replacing solenoid valves in the Post Accident Sampling System (PASS) and the RHR Service Water System with those qualified for postulated environmental conditions following a LOCA. A Yankee Atomic Electric Company memorandum (VYS 45184, dated March 12, 1984) regarding RHR SW Pump Motor Thrust bearings stated that during the environmental qualification program review it was determined that (i) the motor cooling water lines were originally classified incorrectly as non-nuclear safety versus safety class 3; and, (ii) the cooling water lines should be designed to withstand seismic events.

The inspector questioned the licensee regarding the operability of the RHR service water system and whether this finding was reportable to the NRC. The licensee stated that an engineering review had been performed, but not documented, which concluded that the RHR Service Water pump motor bearing cooling lines were adequately supported to withstand seismic events even though not designed for this purpose. The licensee further stated that the safety classification would be changed and the seismic evaluation documented. The inspector will follow the licensee's actions in a future inspection (UNR 84-12-03).

The inspector observed the installation of 4 solenoid valves for the PASS and 3 for the RHR Service Water system and verified that the nameplate model numbers and voltage ratings were as specified in the design documents. During discussions with Operations personnel, the inspector noted that the licensee had identified that the three-way solenoid valve installed in the RHR Service Water cooling line did not shut fully upon securing the pumps. Licensee representatives evaluated the cause and possible corrective actions while operating the cooling line in a continuous bypass mode.

The licensee subsequently concluded that the new qualified switches were subject to failure due to sediment in the service water lines and re-installed the original, unqualified solenoid valves in accordance with a revision to the EDCR. The licensee concluded that subsequent plant operation with the original valves could be justified technically and a Justification for Continued Operation was prepared. The inspector will follow the licensee's resolution of this item (UNR 84-12-04).

No violations were identified during the review of plant modifications.

8. 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 4113.01, MSIV Partial Closure Test, June 15, 1984
-- OP ·102.02, Two Rod Interlock Functional Test, June 21, 1984
-- OP 4102.01, Functional Test of Refueling Interlocks, June 21, 1984

No violations were identified.

9. Maintenance Activities

Maintenance activities associated with the replacement of the UPS batteries were reviewed on July 4, 1984 to assure that the requirements of AP 0021 were met, qualified replacement parts were used, administrative approvals and tagouts were proper.

Work was done under MR 84-812 and 813 for battery A and B, respectively. The existing UPS batteries were replaced with new Exide Series E cells due to a copper contamination and terminal seal problem with the old cells. Refer to paragraph 2.b above for further information on this item.

The licensee prepared a one time procedure to complete the work: OP 5200.27, Installation Guideline For UPS Battery Replacement. The inspector noted during the review of work activities that the NEPSCO workers had a draft copy of OP 5200.27 for reference during the removal of the old batteries, which were to be discarded. Use of the unapproved procedure was discussed with the work party leader and the Maintenance Department Senior Engineer. The workers knew that work could proceed up to the point in the procedure (Step 12) that accomplished removal of the old batteries. Work beyond that point would have to be done with an approved procedure. Removal of the old cells can be considered a task that is within the skills of the workers and, as such, could have been accomplished without any procedure.

The inspector acknowledged the above, but expressed his concerns regarding the poor example that could be established by authorizing work under unapproved procedures.

Procedure OP 5200-27 was approved on July 9, 1984 and the new UPS batteries were installed. The inspector had no further comments on this item. The completion of maintenance activities in accordance with established administrative controls will be reviewed further on subsequent routine inspections.

No violations were identified.

10. Review of Plant Evolutions, Trips and Events

Events that occurred during the inspection were reviewed 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 review of plant trips and operational events:

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

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

-- 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.

Events reviewed during this period included an unshielded traversing incore probe (TIP) on June 15, 1984 and an inadvertent scram signal that was generated while the reactor was shutdown.

a. Unshielded Traversing Incore Probe

During the performance of routine core flux mapping with the reactor at 92% full power on June 15, 1984, the drive mechanism for TIP #1 malfunctioned at 9:05 A.M. and retracted the detector beyond the shielded position and into the drive housing. The probe had made one pass through the reactor. The unshielded probe generated excessive general area dose rates in the Northwest corner of the Reactor Building 252 ft. elevation.

Local radiation readings increased to about 5 R/hr general area, with levels on contact with the drive housing reaching 100 R/hr. An area radiation monitor detected the elevated radiation levels and alarmed the condition in the control room. Workers installing cables in the general vicinity of the drive left the area when the local high radiation alarm sounded. Health physics technicians responded to the area and confirmed the elevated readings using portable (teletector) monitors.

Plant operators declared an Alert at 9:18 A.M. in accordance with the emergency plan due to unexpected increases in radiation levels of greater than 1000 times normal. The three states were notified of the detector problem and the stable plant conditions. NRC Region I was notified of the event by telephone from the Resident Inspector at 9:25 A.M. The licensee made notifications to the NRC Duty Officer using commercial lines at 9:30 A.M. since the ENS line was out of service at the time. The resident inspector monitored the licensee's response to the event and maintained periodic contact with NRC Region I management until the emergency condition was terminated.

The Reactor Building was evacuated and guards were posted to establish positive access controls. Personnel accountability procedures were completed at 9:49 A.M. The TLDs for nine people working in the vicinity of the drive housing were pulled and read for exposure readings. No excessive exposures occurred. The pocket dosimeters for the workers showed no exposures in excess of 80 mRem for the morning work activity. There were no releases of radioactive material inside the plant or offsite.

The licensee obtained assistance from his engineering organization and reviewed alternative contingency plans to recover from the event. Radiation decay curves for the probes showed that the dose rate would decrease by a factor of 100 in about 24 hours. The licensee determined that the best approach would be to control access to the area and wait until the detector had decayed prior to attempting to place it back into its shield. The electrical power to the other two detector drives was de-energized to preclude any other drive malfunctions.

The licensee de-escalated from the Alert condition to a non-emergency status at 1:19 P.M. since the radiation source was known, decreasing and controlled. The licensee completed confirmatory radiation surveys in areas adjacent to the TIP drives to establish the boundaries of the high radiation areas. Access to the Reactor Building was permitted at 2:10 P.M. after a steel barricade was constructed to provide access control to the drive area. The licensee deferred detector recovery activities for 24 hours to allow the gamma-tipped detectors time to decay. The licensee began a controlled reactor shutdown at 4:15 P.M. to cold conditions to commence the 1984 refueling outage.

Contact radiation levels on the probe decreased to 6 R/hr as of 10:00 A.M. on June 16, 1984. Radiation levels around the #1 TIP drive were less than 600 mR/hr at 18 inches. Locked High Radiation Area controls for the area were relaxed at 10:00 A.M. I&C technicians cut the gamma-tip detector from the drive at 1:30 A.M. on June 17, 1984 and transferred the detector to a shielded drum for storage. Contact dose rates on the TIP were measured at 3 R/hr. The highest whole body dose received by an individual for the operation was reported to be less than about 30 mRem. Dose rates on the #1 TIP drive housing decreased to less than 7 mR/hr.

A new gamma-tip detector was installed for the #1 TIP drive during the outage. The licensee identified the cause for the drive malfunction by June 20, 1984. The inadvertent withdrawal of the TIP occurred when a shaft key between a hobbed wheel and the load shaft in the motor drive train fell out and allowed the spring-loaded Gleason wheel assembly to fully retract the TIP into the drive housing. This failure mode bypassed the fail safe features of the drive motor control circuitry, that were designed to stop the detector when it reaches the shield position. A different type of shaft key was installed on the drive shafts for all three TIP machines to preclude a recurrence of the June 15, 1984 failure.

The licensee reported this event to the NRC as LER 84-07 dated July 12, 1984.

No violations were identified.

b. Inadvertent Scram Signal on June 17, 1984

With the reactor in cold shutdown at 7:56 A.M. on June 17, 1984, an inadvertent scram signal was generated during the performance of a level instrument surveillance test, when a valving error by I&C technicians caused an erroneous low vessel water level signal to occur. Control rod friction testing was in progress at the time. Plant operators reset the scram after verifying the cause and testing resumed. A CFR 50.72 report was made to the NRC Duty Officer at 9:05 A.M.

The incident occurred during the performance of OP 4313 on level instruments LT 2-3-57A/B and 58A/B. During the test, technicians inadvertently opened the low side supply valve for two transmitters, which provide input to reactor vessel low level trip channels C and D. This caused a full scram to occur. The control room operators reset the scram after verifying the cause for the trip signal.

The licensee reported this event to the NRC as LER 84-09 dated July 17, 1984.

No violations were identified.

11. Refueling Activities

Plans and procedures established by the licensee in preparation for refueling operations were reviewed to verify compliance with regulatory requirements. Plant systems status was observed and completed test procedures were reviewed to verify that required surevillances were completed. Refueling activities in progress were observed for compliance with established procedures and regulations.

a. Reload License Submittal

The inspector reviewed the 1984 Refuel Outage Plan that provided the licensee's planning guide for overall outage schedule, work lists, design change and modification packages and required surveillance. No inadequacies were identified.

The inspector reviewed the licensee's letter FVY 84-50 dated May 17, 1984 to NRR regarding the Cycle 11 core performance analysis. The analysis found that the margins of safety contained in the limiting conditions for operation in the current technical specifications were not reduced. The licensee determined pursuant to 10 CFR 50.59 that there was no need to submit a change to the technical specifications. The inspector noted that the core (MCPR operating) limits contained in Attachment Table A.1 of FVY 84-50 were no more restrictive than the current limits in Technical Specification Table 3.11-2.

The Plant Operations Review Committee (PORC) reviewed the Cycle 11 analysis results in meeting 84-12 on May 4, 1984. PORC accepted the results, contingent upon satisfactory resolution of the comments documented in a May 8, 1984 memorandum to the Operations Supervisor. PORC noted that the engineering organization (YNSD) did not analyze for the special testing configuration under natural circulation conditions, and thus, could not address whether the existing MCPR operating limits were still applicable for that condition. Although the licensee has no plans to repeat the special testing completed in 1982, the current technical specifications do not specifically exclude re-entry into that operational condition. The NRC's safety evaluation does indicate, however, that the special testing allowance applied only to a limited period during Cycle 8 operations.

PORC recommended that the following actions be taken: (i) the VY Manager of Operations (MOO) issue a directive that the plant be administratively restricted from operating in the special stability testing configuration; and; (ii) the special testing MCPR operating limits be removed during a subsequent change to the technical specifications. The licensee tracked this issue as a PORC follow item. The issuance of a MOO Directive regarding special stability testing will be followed on a future inspection (IFI 84-12-05).

No violations were identified.

b. Refueling Procedures

The following procedures were reviewed in preparation for witnessing the indicated activities. The procedures were approved in accordance with the licensee's administrative requirements and were found to be technically acceptable to accomplish the intended task.

AP 1000, Refueling, Revision 9, October 3, 1983
OP 1100, Refuel Platform Operation, Revision 11, March 3, 1984
OP 1101, Fuel Assembly Movement, Revision 11, April 16, 1984
OP 1111, Control Rod Removal and Installation, Revision 11, July 6, 1982

OP 1403, Fuel Bundle Nondestructive Testing/Reconstitution, Revision 5, March 16, 1983
OP 1408, LPRM Removal and Replacement, Revision 6, December 27, 1983
OP 1410, Fuel Loading, Revision 11, May 12, 1983
OP 1411, Core Verification, Revision 6, December 27, 1983
RP 1620, Fuel Sipping, Revision 8, December 27, 1983
RP 1621, Vacuum Fuel Sipping, Revision 0, June 20, 1984

No violations were identified.

c. Refueling Prerequisites and Periodic Tests

The inspector reviewed test results, completed check lists and logs to verify that refueling prerequisites established by procedures and the technical specifications were completed as required. The following items were reviewed during the period of June 20-23, 1984.

(1) The requirements of Technical Specification 3.12 governing refueling interlocks, core monitoring, spent fuel pool water level, reactor mode switch position, fuel movement, spent fuel pool temperatures, SRM coincidence logic and reactor building integrity;

(2) The requirements of Technical Specification 3.5.H.4, governing core spray, residual heat removal, diesel generator and condensate storage tank operability; and,

(3) The completion of the following prerequisite and periodic tests: SRM Response checks, completed per VYOPF 4420.01; refueling outage tests completed per OP 4102; and, refueling prerequisites completed per OP 1410.

No violations were identified.

d. Refuel Operations

Fuel handling and associated activities were observed during the period from June 20-26, 1984. The inspector witnessed the movement of fuel bundles LY 4841, LJP 209 and LJP 210 on June 22, 1984. Activities were monitored in the control room and on the refuel floor during both day shifts and back shifts. The following items were verified during inspector observations.

(1) approved procedures governing the activity were followed and applicable prerequisites were established and maintained;

(2) fuel transfer status boards were established and maintained in the control room and on the refuel floor;

(3) applicable procedures for SNM accountability were in use, followed and maintained;

(4) communications were maintained between the refuel floor and the control room;

(5) source range monitors responded as expected during fuel transfers and were being used by shift personnel to monitor core geometry changes;

(6) control room and refuel floor staffing met procedural and regulatory requirements; and,

(7) health physics and housekeeping controls, including control of loose objects over the reactor cavity and the spent fuel pool, were established and maintained.

No violations were identified.

e. Spent Fuel Pool Activities

The inspector noted through procedure reviews and direct observation at various times during the inspection period that the following controls were established for activities associated with the spent fuel pool:

(1) water level and temperature were maintained as required;

(2) reactor building integrity was maintained;

(3) the spent fuel pool cooling system was operable;

(4) process and area radiation monitors were operable; and,

(5) fuel pool clarity was acceptable.

No violations were identified.

f. Fuel Sipping

The licensee sipped fuel to identify the suspected leaky bundle. Bundle LJU 720 was found to have one failed pin. LJU 720 was one of two GE lead test assemblies that operated in cycle 10 and was discharged with a 23,000 MWD/ST exposure. The licensee completed detailed examination of LJU 720 and sister bundle LJU 719 in July 1984 to document the as-found conditions. The licensee's actions to either replace or reconstitute LJU 720, and to determine the bundle failure mechanism will be reviewed on a future inspection. This item is being tracked as inspection item 84-10-01. The inspector had no further comment on this item at the present time.

No violations were identified.

g. Dropped Fuel Pin

During fuel bundle inspection activities at 1:00 A.M. on July 19, 1984, a single spent fuel pin was dropped from the handling tool as the pin was lifted from the bundle for examination. The pin landed undamaged on the spent fuel racks. There was no release of radioactive material. The pin dropped because of a lack of proper engagement with the handling tool. The pin was recovered using standard handling tools and placed in a pin storage bin. The pin will be returned to the fuel bundle. The bundle was not scheduled for subsequent use in the reactor.

No violations were identified.

12. Core Spray Valve Isolation Logic

A potentially generic issue was identified in licensee event report (LER) 50-387/84-26 dated June 14, 1984, regarding the core spray (CS) system full flow test valve isolation logic. LER 84-26 was reviewed in conjunction with Final Safety Analysis Report (FSAR) Figure 7.4-5, Technical Specification Table 3.2.1 and Control Wiring Diagram 191301, Sheets 1150, 1156 and 1163 to determine whether the problem described in the LER was applicable to Vermont Yankee.

The VY as built configuration was found to agree with the design described in the FSAR functional control diagrams. The CS full flow test valve will close in response to either a high drywell pressure condition, or a low-low reactor vessel water level condition, coincident with low reactor pressure, as required by the design specifications. The inspector had no further questions on this item.

No violations were identified.

13. 1984 Annual Emergency Exercise

The licensee conducted a critique of his response to the June 15, 1984 Alert emergency during a meeting on June 23, 1984. The inspector attended the critique session as an observer and noted that the licensee completed an objective evaluation of his emergency preparedness response actions. The inspector summarized his comments on the licensee's response after the licensee completed his critique.

The inspector identified no concerns that had been identified by the licensee. Within the scope of the inspector's observations from the control room, the technical support center and the operations support center during the emergency, the inspector concluded that the licensee performed well to: (i) identify the nature and cause of the high radiation condition and take immediate actions to protect personnel; (ii) classify the event, notify the NRC and the states, and activate his emergency response centers; (iii) use the emergency response centers and resources to evaluate the problem and determine the best course of action; and (iv) recover and de-escalate from the emergency condition.

The licensee completed a review of his response to the emergency condition and asked the inspector whether the NRC would consider giving credit for the event as fulfilling the requirements of the 1984 annual emergency plan exercise scheduled for November 14, 1984. This matter was discussed with NRC Region I management. The inspector informed the licensee that the staff would not likely accept the event as a substitute for a planned exercise, due to the exercise controls (controller, observers, etc.) that were not available on June 15, 1984 and which are normally utilized during drills to effectively evaluate response activities. The major limitation of the June 23, 1984 critique was the lack of stationed observers during the event, who would have had no role other than to evaluate response actions.

However, the inspector informed the licensee that the NRC staff would formally respond to a written request for exemption from the requirements of 10 CFR 50, Appendix E which addresses the justifications for accepting the event in lieu of the scheduled exercise.

The licensee noted the inspector's comments and stated his intention to submit an exemption request. This item will be reviewed further on a future inspection pending receipt of written communication from the licensee.

14. Inservice Inspection Activities

a. Feedwater Nozzle Examinations

The inspector reviewed the NDE inspector qualification tests that were completed on June 23, 1984 in preparation for the examinations to be completed on the four reactor vessel feedwater nozzles. The qualification tests were completed on a vessel nozzle mockup that had been constructed with defects. The size and location of the defects were described on Drawing No. E-7983602. The mockup contained several geometry and flaw indications that had to be identified to successfully complete the test. The results from the qualification tests completed on June 23, 1984 were reviewed against the drawing. The individuals performing the examinations were successful in identifying the required number of defects.

No rejectable indications were identified on the four feedwater nozzles during the subsequent examinations.

No violations were identified.

b. Information Notice 84-41

The NRC issued Information Notice (IN) 84-41 on June 1, 1984 as a result of industry experiences with intergrannular stress corrosion cracking in BWR plants. The inspector noted that the licensee had received the notice and reviewed it for applicability to VY. The licensee examined the problems discussed in the notice as they applied to VY and determined that no augmented inspections were warranted during the current outage. The licensee's conclusions were identified to NRC:NRR in letter FVY 84-77 dated July 3, 1984.

No violations were identified.

c. Recirculation System Weld Examinations

The licensee informed the inspector that a flaw was identified on weld number 26A as a result of the augmented inservice inspections completed on the recirculation system as of July 6, 1984. Joint number 26A is the elbow to safe end weld for nozzle N1B on the 28 inch diameter suction line to the 'B' recirculation pump. The indications were circumferential and consisted of two segments each about 10 inches in length. Evaluations were completed to fully characterize the flaw.

As of July 6, 1984, all 48 welds in the original sample selected for examinations were scanned at least once. Of these 48, 15 welds were found free of flaws, and 10 of these 15 were welds that were overlay repaired during the 1983 outage. The overlayed welds were inspected for weld metal integrity and clad bonding to the base metal only. Additional scans on selected welds were completed to assist in the evaluation of the joints.

Five additional welds were added to the list to be examined as a result of the finding for weld 26A. Additionally, the licensee added five more sweepolet welds on the recirculation ring header to meet the requirements of NRR's generic letter 84-11. This brought the total number of welds to be examined to 58 as of July 6, 1984.

The licensee reported on July 16, 1984 that RHR system weld #32-4 was found to have an axial flaw in excess of 10% of the through wall dimensions. RHR line #32 is a part of the 20 inch diameter suction line to the RHR pumps for shutdown cooling operations. Weld #32 is located inside the drywell. Two additional RHR system welds were added to the sample of welds selected for examination.

The licensee completed overlay repairs for weld #32-4. An overlay repair was also completed on weld #32, which had a min-overlay applied during the 1983 outage. Weld joints with circumferential indications were found adequate for another cycle of operation without overlay repair.

The final results of the 1984 augmented examination program for recirculation system piping were discussed in a meeting with the NRC staff on August 2, 1984. The results of the Staff review of the 1984 program and the additional corrective actions to be taken by the licensee will be reviewed on a future inspection (IFI 84-12-06).

No violations were identified.

15. Medical Screening for Contractor Personnel

a. Worker Complaint

A worker came to the Resident Office at 10:15 A.M. on June 28, 1984 to register a complaint regarding his termination that day for failing a medical screening exam that was taken on June 12, 1984. The worker stated that 13 other workers were also terminated that day due to medical screening results. The individual was a non-permanent employee with the New England Power Service Company (NEPSCO), a contractor for the licensee. The worker alleged that the licensee knew of the medical results since the exam was taken on June 12, 1984, but used him inside the drywell until the job was finished.

The worker had been on site since June 13, 1984 and assigned to work in the drywell to remove piping insulation. He wore respiratory protection equipment during this work. The worker stated that he received exposures up to the weekly limit of 600 mRem during the weeks of June 18 and 25, and was upgraded to an 800 mRem weekly limit on June 27, 1984.

The worker had prior medical clearance for radiation work, but was sent to NEPSCO's Westborough Office on June 12, 1984 to undergo further medical examination prior to reporting for work at the Vermont Yankee site. The worker provided a copy of a June 26, 1984 letter from NEPSCO's physician which he received on June 27, 1984. The letter reported the results of a physical examination and provided the reason why approval for employment as a radiation worker was rescinded. The worker believed that his weekly exposure should not have been upgraded to 800 mRem on June 27, 1984, given the examination was taken on June 12, 1984 and the results were known by June 26, 1984.

The inspector informed the worker that the matter would be reviewed and that he would be notified of the inspection results.

An inspection was conducted in this area to determine:

(1) whether the licensee knew of the medical results and deferred action until certain work was completed inside the drywell;

(2) why workers were allowed to start radiation work prior to having the final results from the medical screening tests; and,

(3) whether the apparent problems in processing NEPSCO personnel occurred for other contractor groups.

The inspection results are summarized below. The information below provides the basis for a conclusion that there was no attempt by the licensee to knowingly expose workers to radiation who were not medically qualified for radiation work.

b. Findings

The following information was obtained during interviews with the plant nurse and other licensee employees, and through reviews of medical information sent to the licensee by NEPSCO and other contractors. Health physics records were also reviewed. NEPSCO representatives were contacted as necessary by the plant nurse to provide information.

The inspector began this review on June 28, 1984 without disclosing to the licensee that an allegation had been received. On June 29, 1984, the licensee terminated a second group of NEPSCO workers (18 personnel) due to adverse medical screening results.

Health physics records were reviewed for the period from June 13-28, 1984 to identify all work performed by the recently terminated NEPSCO workers under radiation work permits that required the use of respirators. This review determined that at least half of the NEPSCO workers performed work in respirators inside the drywell to remove piping insulation during the period from June 18-27, 1984. Based on a review of drywell survey results and interviews with HP technicians cognizant of the drywell work, the inspector determined that respirators were used in the job as a precaution during insulation removal only. Airborne levels in the drywell were sufficiently low (ranging from 5.0 E-10 to 5.0 E-9 uCi/cc, with one reading of 1.0 E-8 uCi/cc for a brief period) that respiratory protection factors were not needed or taken for the work.

Based on the above, there were no safety concerns or a significant health issue for the NEPSCO workers, since there were no elevated air concentrations in the drywell during the work period of interest. There were no violations of 10 CFR Part 20 requirements.

Approvals to increase the weekly administrative exposure limits for NEPSCO personnel during the period from June 13-28, 1984 were reviewed. The upgrades were processed and approved in accordance with the established procedure and the increased dose intervals were reasonable for the insulation work being done in the drywell. There was no evidence that excessive increases in the dose limits were approved to "burn-out" the workers prior to terminating them from the job.

NEPSCO performs annual medical exams for its employees and provides a monthly status listing of personnel who have been medically approved for radiation work. A copy of the status report for June, 1984 was reviewed. There were about 280 NEPSCO personnel on the June 1984 status report. About half of this number are assigned to Vermont Yankee for outage related work.

The NEPSCO workers terminated on June 28, 1984, were on the list as having medical approval for radiation work. This medical approval was given based on exams conducted under the licensee's old administrative requirements. The old requirements stipulated that a worker be evaluated through a physical exam, a pulmonary function test and a CBC blood analysis prior to being approved for work with respiratory protection equipment.

The licensee augmented his administrative controls on June 1, 1984 through a revision to AP 0800, which imposed more restrictive medical requirements for work in the Radiation Controlled Area (RCA). AP 0800 required that personnel whose work could expose them to radiation in excess of 250 mRem per quarter be medically screened prior to being assigned work in the RCA. The new AP 0800 requirements met the evaluation criteria listed in section 7.4 of NUREG 0041, plus incorporated criteria that could be used in screenings for drug and alcohol abuse. Letters were sent to NEPSCO and other contractors to advise them of the new requirements and their responsibility to ensure that employees were medically evaluated under the AP 0800 requirements.

NEPSCO was initially given a June 1, 1984 deadline to complete augmented physical exams for its employees under the new procedure. NEPSCO began re-examining workers to meet the new requirements. As medical evaluations were completed, the NEPSCO physician reported the results in memoranda, which were faxed to the VY site nurse for processing and became supplements to the June status report. The supplements identified individuals that were either approved for work under the new requirements, or, for whom medical approval was rescinded on the basis of the additional evaluations. The June status listing was updated almost daily from June 13-28, 1984. Most supplements addressed one or two individuals. In the case of the NEPSCO workers terminated on June 28, 1984, medical approval was rescinded in two supplements dated June 26 and 27, 1984.

The initial deadline to complete the augmented evaluations was extended until June 15, 1984 since medical exams were still in progress. The refueling outage began on June 15, 1984 and the deadline for completing the physicals was extended further while NEPSCO workers were processed in under the old medical approvals. Plant management approved the exceptions to the AP 0800 requirements on the basis that NEPSCO personnel had medical clearances under the old requirements, the re-examinations were in progress, and personnel who did not meet the new requirements could be handled on a case by case basis as they arose. When the supplement for June 26, 1984 was received with the large number of workers who did not meet the medical requirements, plant management rescinded the extensions given to NEPSCO and terminated all workers at the site who were not approved under the new requirements.

The workers were relieved from work assignments and processed off the site by June 28, 1984. The medical approvals were acted on within 1 to 2 working days once the information reached the site. The biggest delay in acting on the medical results (from June 12-26) was attributable to processing time at NEPSCO. This period includes the time for a laboratory to complete blood tests and report the results to the physician. Further, it is probable that the processing delays from June 12-26, 1984 were due in part to the large number of NEPSCO workers being evaluated under the new requirements.

The inspector reviewed medical approvals on file for other personnel. A random sample of medical results was reviewed for about 20 individuals employed by four other contractor organizations at the site. No discrepancies were identified.

No violations were identified.

16. Management Meetings

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

Unresolved items are items for which further information is required to determine whether the items are acceptable or violations. Unresolved items are discussed in paragraphs 6 and 7 of this report.