U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 87-09

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

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

Facility: Vermont Yankee Nuclear Power Station

Location: Vernon, Vermont

Dates: April 7 - June 1, 1987

Inspectors: William J. Raymond, Senior Resident Inspector
Donald R. Haverkamp, Project Engineer

Approved by: Thomas C. Elsasser 6/23/87
Thomas C. Elsasser, Chief, Reactor Projects Section 3C date

Inspection Summary: Inspection on April 7 - June 1, 1987 (Report No. 50-271/87-09)

Areas Inspected: Routine, unannounced inspection on day time and backshifts by the resident inspector of: actions on previous inspection findings; physical security; plant operations; licensee event report 87-02; actions in response to the discovery of contaminated tools outside the radiation controlled area; maintenance activities; dryer storage pit wall weepage; surveillance activities; leak rate testing of torus penetrations; fuel handling activities; submission of the semiannual effluent release report; plans and procedures to install a single high density storage rack in the spent fuel pool; and, preparations to perform RHR pump wear ring inspections and replacement. The inspection involved 228 hours.

Results: No violations were identified. Routine reviews of plant activities identified no conditions adverse to safe plant operations. A licensee identified violation (not cited) concerned the inadvertent release of contaminated tools outside the radiation controlled area (Section 5.6). While the specific contamination incident was not significant, it appears as a recurrent problem in the control of low level radioactive materials.

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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. P. Donnelly, Maintenance Superintendent
Mr. R. Lopriore, Maintenance Supervisor
Mr. J. McCarthy, ALARA Engineer
Mr. R. Pagodin, Technical Services Superintendent
Mr. J. Pelletier, Plant Manager
Mr. J. Sinclair, Plant Administrative Supervisor
Mr. R. Wanczyk, Operations Superintendent

2. Summary of Facility Activities

The plant continued routine operations at rated power during the inspection period, and began a power coastdown on May 18, 1987 for end-of-cycle operations. NRC Region I specialist inspectors completed reviews during the period of April 13-24, 1987 of the emergency preparedness program (Inspection 87-07), and the maintenance program (Inspection 87-08). The licensee completed action during the period to inspect the Bingham RHR pump wear rings for IGSCC cracking and to rebuild the pumps with new impellers having integral wear rings.

3. Status of Previous Inspection Findings

3.1 (Open) Violation 86-25-01: Failure to Meet ASME Section XI Requirements for Inservice Testing. The licensee responded to this item by letter FVY 87-30 dated March 12, 1987 and requested that the NRC staff reassess the violation based on the information provided in the response. The inspector reviewed the response and noted that no information was provided which demonstrated that the actions taken following the testing of the "B" core spray pump on October 7 and November 25, 1987 met the requirements of Subsection IWP 3230 of the Section XI Code, Winter Addenda, 1980 Edition for inservice testing (IST). In his response, the licensee presented his interpretation of a method of implementation of the code requirements; however, the licensee's position is inconsistent with the NRC staff position as established in a March 17, 1980 memorandum from IE:HQ, as described on page 10 of Inspection Report 86-25.

The licensee stated that if test data is found in the "required action" range, then the shift supervisor would be responsible to develop an action plan to complete an evaluation of the operability of the pump within 96 hours from when the initial data was obtained. The licensee based this position on the provisions of IWP 3230(d) which states that "when tests show deviations greater than allowed, the instruments involved may be recalibrated and the test rerun".

The inspector noted that the intent of the code was to allow the licensee to discredit a data set if the instruments used to gather the data are suspected of being faulty. In this case, the approach should be to disregard the data and rerun the test with known good instruments that are in calibration. This is a reasonable approach when (or if), as the licensee indicated, "the shift supervisor has reason to believe the data taken did not represent the true condition of the pump for some reason, such as transcription error, instrument malfunction, operator error in using or reading the instrument". However, given that the prerequisite for each test run is that the measurements be made by qualified personnel with good instruments that are in calibration, then to discredit a test run the shift supervisor must have solid evidence that the measurement was faulty and should not invoke the provisions of IWP 3230(d) without due cause.

For the core spray pump testing conducted in the Fall of 1986, the operators who took data were qualified and used good equipment that was "in calibration". The instrumentation repeatedly showed that the "B" core spray pump had a vibration problem that was classified at the "alert" or "required action" level based on the baseline vibration for the pump. The licensee's evaluation of the "status" of the pump did not involve recalibration of the IST instruments and rerunning the test. Instead, considerable effort was expended using maintenance personnel and consultants with sophisticated vibration spectrum analysis equipment to ultimately determine that the pump was in good operating condition (except for the upper motor bearing), and that the IST vibration measurement equipment was too sensitive and not well suited for the program. This level of effort went beyond the followup action suggested in IWP 3230(d) to "recalibrate and rerun with the IST measurement equipment". This effort is more appropriately considered as part of a corrective action plan per IWP 3230(c), which requires that the pump be either "replaced, repaired, or an analysis done to demonstrate that the condition does not impair the pump operability and that the pump will still fulfill its function". This was the ultimate conclusion for the core spray pump.

However, the success of this outcome cannot be known a priori. The Section XI code, instead of requiring the use of highly sophisticated vibration spectrum analyzer equipment, allows the use of much simpler, highly portable equipment by qualified operators to monitor vibration levels as a general indicator of the performance of operating equipment. This program is sufficient to detect the onset of trends or conditions that could jeopardize pump operability. The intent of the code is that should the IST measurements detect vibration levels at the "alert" range, actions should be taken to address the problem as a maintenance item per IWP 3230(a) and "double the test frequency until the cause of the condition is determined and the condition corrected". Similarly, should the

IST measurements find vibrations to be in the "required action" range, and absent sound evidence that the test data just taken was faulty, then the prudent and required action per IWP 3230(c) is to declare the pump inoperable and to not return it to service until the condition is corrected.

Based on the above, the licensee's response and corrective actions were lacking in that they do not satisfy the explicit requirements or intent of the Section XI code. The licensee should revise his test controls and procedures according to the NRC staff position given above. In this approach, when IST test data are taken, they must be analyzed within 96 hours of the measurement as required by IWP 3220. If the data fall within the "required action" range, and absent justification to invoke the provisions of IWP 3230(d), the pump should be declared inoperable. The operator should then follow the provisions of any technical specification action statement that applies.

This item was discussed in detail with the Operations Superintendent and other members of the licensee's engineering staff during the inspection. No consensus was achieved regarding acceptance and implementation of the NRC staff position. The licensee requested the inspector to review the matter further with NRR personnel and to determine specifically whether the provisions of IWP 1500 would allow "further testing when deviations are identified" prior to declaring the subject equipment inoperable. This item remains open pending further review of the licensee's position to determine what additional actions may be necessary beyond those documented in FVY 87-30.

3.2 (Open) Unresolved Item 87-02-03: Core Spray Safe-End Inspections. By letter dated May 28, 1987, NRC:NRR found the licensee's plans to inspect the repaired core spray safe-ends during the 1987 refueling outage, instead of replacing them, to be acceptable for one cycle of operations, provided the inspection results are satisfactory. The licensee is to provide a description of the inspection plans to NRC:NRR for review and the results of the inspections will be reported within three weeks of the startup from the refueling outage.

The inspector had no further questions on this item at the present time. This item remains unresolved pending completion of the safe-end inspections during the 1987 refueling outage and subsequent NRC review of the results.

3.3 (Open) Unresolved Item 86-10-02: CST Leakage Monitoring and Repair. This item was last reviewed during inspection IR 86-22, and is discussed further in section 11.1 below. The licensee completed ultrasonic wall thickness measurements on the tank floor and determined that, since no further degradation has occurred, no actions to repair the tank will be required during the 1987 refueling outage. Based on the above inspection results and the ongoing leakage monitoring program, it appears that there has been no further corrosion degradation of the tank floor.

The licensee stated that a service request will be issued to YNSD to provide an engineering evaluation of what long term corrective actions are appropriate. This item remains open pending completion of the licensee's evaluation to determine the appropriate long term action plan for the CST.

3.4 (Closed) UNR 82-01-03: Requirements for Seismic Monitoring Instruments. Subsequent inspector review of this item determined that the seismic monitoring instrumentation installed at the site meets the requirements specified in Amendment 16 of the Final Safety Analysis Report dated October 23, 1980, and Section 3.4 of the NRC Staff Safety Evaluation dated June 1, 1987. Although no LCO or surveillance requirements are presently stated in the technical specifications, the licensee has and implements procedures DP 4396 and AP 0150 to verify the operability of the strong motion accelerograph, and monitor its status on a periodic basis. The combination of equipment and administrative controls is sufficient to meet the requirements established for License DPR-28. No further actions are required. This item is closed.

3.5 (Closed) Follow Item 82-01-06: Non-Licensed Operator Training. The inspector noted that procedure DP 0160 was revised as expected in the Spring of 1982, but was subsequently cancelled. The non-licensed operator training program requirements were re-issued in procedure AP 0715, Non-licensed Operator Training. The inspector reviewed the training program requirements specified in Revision 7 to AP 0715 dated December 30, 1985 and noted that the program includes classroom lectures on plant systems and procedures. The inspector identified no inadequacies. The inspector noted further that the licensee's non-licensed operator training program has received INPO accreditation. This item is closed.

3.6 (Closed) Unresolved 85-25-04: Familiarity with the Requirements of AP 0020. The implementation of electrical jumpers and lifted leads by licensee personnel was reviewed during subsequent routine inspections, and most recently during this inspection, as discussed in Section 5.7 below. No inadequacies were identified which would indicate a lack of familiarity with the requirements of AP 0020 by licensee personnel. This item is closed.

3.7 (Closed) Unresolved Item 85-40-08: Update LER. The licensee submitted Revision 1 to LER 86-13 by letter dated May 2, 1986 which described the resolution to the support deficiencies noted during the recirculation pipe replacement outage. The resolution of the support discrepancies were reviewed and accepted by the NRC staff during Inspections 86-10 and 86-13. The licensee met his commitments for this item and no further actions are required. This item is closed.

3.8 (Closed) Follow Item 83-02-06: Corrective Actions for Tagging Error. This item involved the review by licensee personnel to determine whether controls for switching and tagging operations needed to be enhanced as a result of a tagging error that caused a service water leak which threatened the plant electrical supply. A review of the tagging controls identified alternatives for enhancing the administration of system tags, as documented in a memorandum to the Operations Superintendent dated August 19, 1983. The recommended action was to use computer based tracking methods to assist the control room operator in the bookkeeping and information retrieval task involved with switching and tagging controls. However, the licensee concluded that the tagging system enhancements should be deferred until the plant computer upgrade is completed, which is scheduled to be done in phases during the 1987 and 1989 outages. There are no actions presently in progress or planned to address tagging controls, pending completion of the plant computer changes and installation of the safety parameter display system.

The inspector noted that the existing administrative controls for switching and tagging operations are adequate. There have been no violations identified during NRC inspections of tagging operations since 1983, and specifically during periods that require extensive tagging operations, such as the 1984 and 1985 outages. The inspector had no further comments on this item at the present time. This item is closed.

3.9 (Closed) Unresolved Item 83-01-07: Corrective Actions for Potential Vessel Drain Line Leakage. No further leakage was noted from the reactor vessel bottom head drain line during subsequent operations. The licensee completed actions during the 1985-1986 refueling outage to replace the vessel bottom head drain line in accordance with the recirculation system design changes made per EDCR 85-01 with SA-106 grade B carbon steel material, which is not susceptible to intergranular stress corrosion cracking. This item is closed.

3.10 (Closed) Follow Item 83-09-04: Closure of Jumper and Lifted Lead 83-149. The licensee issued Job Order File 83-33 for MR 82-1151 to document the wiring modifications made per J/LL 83-149. Since the change involved a one-for-one replacement of terminal strips within penetration X102, a design change did not occur and the item could be processed as maintenance activity. J/LL 83-149 was satisfactorily closed on September 25, 1983. This item is closed.

3.11 (Closed) Violation 83-13-01: Implementation of Wiring Changes for EDCR 82-06. The licensee responded to this violation by letter FVY 83-65 dated June 27, 1983 and completed actions acceptable to correct the identified discrepancies. The actions taken were reviewed at the time the response was issued. However, the response contained technical errors in the description of the wiring changes involved in the EDCR 82-06 modifications. The licensee issued FVY 83-80 dated July 26, 1987 which provided a description of the wiring changes that was technically accurate. This item is closed.

3.12 (Closed) Unresolved Item 86-25-03: Labeling of Safety Class Circuits. Licensee action on this item was addressed in an April 8, 1987 response to the Operating Experience Assessment Coordinator. The licensee determined that actions necessary to correct cross labeling and nomenclature problems at the Reactor Building 280 ft. el. for safety class 1 and 2 circuits will be addressed entirely within the scope of EDCR 86-403, which is scheduled to be implemented during the 1987 refueling outage. The subject design change will reroute ECCS cables connecting racks 25-5 and 25-6 to their associated cable trays as necessary to provide ECCS train separation and fire protection to met 10 CFR 50 Appendix R requirements. The design change will eliminate the crossing of SI and SII cables on the 280 ft. el. The licensee's plans for this item via EDCR 86-403 are adequate to address the inspector's concerns. This item is closed.

3.13 (Closed) Violation 83-27-01: Main Steam Line Setpoint Determination. This item was last reviewed in Inspection 84-08. The inspector reviewed OP 4511, Source Calibration of Process Radiation Monitoring System, Revision 15 dated 3/19/87 and noted that Section "System 1-B" contained adequate instructions for determining main steam line monitor trip setpoints from background radiation levels. This item is closed.

3.14 (Closed) Follow Item 83-27-06: RCIC Operability Without Remote Reset. The inspector noted that the remote feature was incorporated in the RCIC system design and installed as a result of a system enhancement in response to TMI Action Plan Item II.K.3.13 of NUREG 0737. The basis for the NUREG item was to increase reliability of the RCIC system by providing for remote manual resetting of the trip throttle valve, which would preclude the need to dispatch an operator to the RCIC room to reinitiate the system after it isolated in response to a high reactor vessel water level isolation signal. The inspector noted that the NUREG item neither redefined the design basis nor identified a new design base accident for the RCIC system. Thus, the RCIC system would still be considered capable of performing its intended function for the design basis event (recover reactor vessel level following a loss of feedwater operational transient - reference FSAR Sections 4.7 and 14.5.4.3). Based on the above, the RCIC system can be considered operable without the remote reset feature. This item is closed.

3.15 (Open) Unresolved Item 87-04-05: Authorizing Maintenance Work. The inspector noted no instances during the inspection period in which the operations shift supervisor initiated work activity prior to "processing the MR". The inspector noted that the licensee reviewed this item and documented his position on the matter in memoranda dated May 21 and June 3, 1987 to the Shift Supervisors. The licensee determined that actions by the shift supervisor to authorize the release of equipment for maintenance prior to processing the MR was consistent with the authority provided to a senior licensed operator, as given in AP 0150, Responsibilities and Authorities of Operations Department Personnel.

The licensee provided clarification to shift personnel on how to use this authority judiciously and provided examples of the types of unusual plant situations that might warrant such actions. The guidance provided to shift personnel was that the shift supervisor could release plant equipment for maintenance prior to processing an MR to prevent loss of important plant equipment, to prevent personnel injury, or to prevent a plant trip.

This item remains unresolved pending further NRC review of the maintenance activity on subsequent routine inspections to verify licensed activities are conducted in accordance with the administrative controls, inclusive of the guidance to shift personnel.

3.16 (Closed) Violation 87-04-04: Control of Maintenance on the Toxic Gas Monitors. The licensee responded to this item by letter FVY 87-51 dated May 8, 1987 to provide his assessment of the event and the corrective actions to prevent recurrence. The licensee determined that the event occurred because communications between the shift supervisor and the work party leader lacked specificity regarding the intended maintenance activity. The event was reviewed with maintenance and operations personnel to clarify the need for good communications during routine activities. The licensee's corrective actions were reviewed and found to be satisfactory. This item is closed.

3.17 (Closed) Unresolved Item 87-06-01: Proposed Staffing Changes. By letter FVY 87-48 dated April 28, 1987, the licensee submitted Proposed Change No. 138 to the technical specifications to address planned administrative changes, including a reorganization of the Chemistry and Health Physics Department, and a clarification regarding the radiation protection manager position. The licensee implemented the organization changes following NRC staff oral approval of the intended changes. This item is closed.

3.18 (Closed) Unresolved Item 85-40-09: RHR Pump Inspections. This item was last reviewed during inspection 87-02 and is discussed further in section 11.3 below. The licensee completed actions during this inspection period to disassemble all four RHR pumps and to rebuild them with impellers having integral wear rings and a material with a Brinnell hardness number that is less susceptible to IGSCC cracking. This item is closed.

3.19 (Open) Unresolved Item 86-25-02: Bingham Pump Minimum Flow Requirements. This item was last addressed in inspection 87-06 and is discussed further in section 11.3 below. The results of visual examination on the four RHR pump impellers showed evidence of significant pitting and cavitation erosion for the number of operating hours on the pumps, which was estimated by the licensee to be 10,905, 7277, 3045 and 3045 hours for the "A", "C", "B" and "D" pumps, respectively. The licensee concluded that some of the observed erosion was due to suction side recirculation flow patterns. This item remains unresolved pending completion of additional licensee evaluation of the adequacy of the RHR and Core Spray minimum flow line sizes in light of the RHR pump inspection findings.

3.20 (Closed) Unresolved Item 86-15-01: IRM Post Maintenance Testing. The inspector reviewed the status of this item with the Instrument & Control Supervisor and noted that the root cause of the IRM detector problem was identified and procedure changes were planned to allow detection of the types of problems that occurred in July, 1986 prior to plant operations.

The three IRM detectors were inoperable due to failure of a fragile piece of electrical conducting foil used between the detector center conductor and the signal/power conductor attached to the detector. The foil is used in the design to absorb vibration during shipment and installation of the detectors. The integrity of the foil cannot be inspected visually, but can be ascertained by changes in detector breakdown voltage readings, based upon measurements made at the factory, upon receipt at the plant, and after installation in the reactor. The problem with the crossed range-control cables between two IRM channels can be detected by a change in test methodology that inserts the test signal upstream of the pre-amplifier in the detector circuit. The licensee has procedure revisions in progress for OP 4301 and 5307 to address both these items. Licensee actions to enhance the present procedures to allow better preoperational verification of the IRM detector operability are satisfactory.

The adequacy of the licensee's post maintenance test program in general was discussed at the March 27, 1987 SALP Management Meeting and in the licensee's response to the SALP assessment (reference FVY 87-44 dated April 24, 1987). The inspector noted that the generally smooth startup and trouble free operating period following the 1985-1986 outage was indicative of a good post maintenance test program. The adequacy of post maintenance testing will be reviewed further during subsequent routine inspections of maintenance activities. This item is closed.

3.21 (Closed) Violation 84-07-02: SBGTS Test Procedures. The licensee's response to this item was provided by letter FVY 84-57 dated June 1, 1984, and was accepted by the NRC staff by letter dated July 3, 1984. The inspector noted that Standby Gas Treatment System test procedures OP 4116 dated November 12, 1986 and OP 4117 dated March 12, 1986 now contain provisions for supervisory review of the completed test results. This item is closed.

3.22 (Open) Unresolved Item 87-04-03: Release of Material from the RCA. The licensee's followup review of the incident involving the resin found in the dumpster was provided in a memorandum by the Plant Health Physicist dated April 21, 1987. Although the original scenario to explain how the resin was placed in the dumpster was not substantiated, the licensee concluded that the existing controls would be sufficient to preclude recurrence of the incident. The matter was discussed with operations personnel to assure they were cognizant of the proper method for disposing resin and of the potential for resin to lift fixed contamination off of a "clean" surface. No inadequacies were identified in the licensee's followup actions.

This item is discussed further in section 5.6 below, which describes inspector concerns regarding the adequacy of the present licensee controls based on recent findings of additional materials that were slightly contaminated outside the RCA. This item remains unresolved pending further NRC review of the licensee's controls.

3.23 (Open) Unresolved Item 87-06-02: Proposed Spent Fuel Pool Changes. The inspector and NRC Region I personnel continued a review of the licensee's proposed plan to add one NES 18X20 high density storage rack to the spent fuel pool per EDCR 87-405. The results of the staff review of the design change will be provided in a subsequent inspection scheduled for June or July 1987, which will include the input from the resident inspector.

The inspector noted further that the licensee has plans in progress to provide another alternative method to provide additional fuel rack storage capacity in or near the spent fuel pool to preserve full core offload capacity following startup from the 1987 refueling outage. This plan involves the purchase and installation of two PAR racks having 10X10 arrays per EDCR 87-406. This option would provide a total storage capacity of 1890 fuel bundles for the pool, which is within the current license limit of 2000 bundles. No decision has yet been made on placing either an NES rack or PAR racks in the pool.

This item is discussed further in section 13.0 below.

4.0 Observations of Physical Security

Selected aspects of plant physical security were reviewed during regular and backshift hours to verify that controls were in accordance with the security plan and approved procedures. This review included the following security measures: guard staffing; vital and protected area barrier integrity; maintenance of isolation zones; and, implementation of access controls, including authorization, badging, escorting, and searches. No inadequacies were identified, except as discussed below.

The inspector reviewed licensee actions taken in May, 1987 in response to information received regarding a potential security threat. The inspector also toured the plant and reviewed sensitive plant areas to verify security and safeguard controls were adequately maintained. No inadequacies were identified.

The inspector reviewed the licensee's response to security events on April 24, May 1, May 15, May 18 and June 1, 1987 involving a moderate loss of security effectiveness. The inspector verified that compensatory measures taken by security personnel were prompt and appropriate for each event.

The inspector reviewed Security Event Report Nos. 87-06 through 87-10 related to the aforementioned events. The inspector verified that the reports accurately described the events and the actions taken by the licensee. No inadequacies were identified.

5.0 Operational Status Reviews

Plant tours were conducted routinely to review activities in progress and to verify compliance with regulatory and administrative requirements. Tours of accessible plant areas included the control room, reactor building, turbine building, diesel generator rooms, and the protected area. Radiation controls were reviewed in areas toured to verify access control barriers, postings and radiological controls were appropriate. Plant housekeeping conditions and shift staffing were reviewed. Shift logs and records were reviewed to determine the status of plant conditions and the changes in operational status.

Plant staffing was reviewed during normal and backshift hours to verify administrative and regulatory requirements were satisfied. Shift staffing was also verified on the swing and mid-shifts on April 14-15, 1987 to verify control room protocol was maintained and that shift personnel were attentive to licensed duties. The inspector also verified shift staffing was adequate to meet technical specification on shift license requirements on April 7, 1987 when an operator was relieved from duty at 2:20 A.M. due to illness. A replacement operator was on site within one hour to fill the AO position. No inadequacies were identified.

5.1 Safety System Review

The residual heat removal, core spray, residual heat removal service water, high pressure coolant injection, service water, reactor core isolation cooling, standby liquid control and standby gas treatment systems were reviewed to verify the systems were properly aligned and fully operational in the standby mode. The review included verification that (i) accessible major flow path valves were correctly positioned; (ii) power supplies were energized, (iii) lubrication and component cooling was proper, and (iv) components were operable based on a visual inspection of equipment for leakage and general conditions. No inadequacies were identified.

5.2 Feedwater Leak Detection System Status

The inspector reviewed the feedwater leakage detection system and the monthly performance summary provided by the licensee in accordance with letter FVY 82-105. The licensee reported that, based on the leakage monitoring data reported as of April 17, 1987, there were no deviations in excess of 0.10 from the steady state value of normalized thermocouple readings, and no failures in the 16 thermocouples installed on the 4 feedwater nozzles. No unacceptable conditions were identified.

5.3 Inadvertent PCIS Group III Isolation

An inadvertent PCIS Group III Isolation occurred at 9:59 a.m. on May 1, 1987 due to a health physics technician error while performing a monthly functional test of the Reactor Building Ventilation Detectors per OP 4511.0. The standby gas treatment system automatically initiated and the normal reactor building ventilation system isolated as required. All plant systems responded appropriately. Plant operators took actions to reset the isolation and return systems to a normal status.

The actuation occurred during the calibration when PCIS relay 16A-K49B was bypassed to allow testing of the North reactor building ventilation detector, 17-452B, and the field technician exposed the South detector (17-452A) to the calibration source. The licensee's subsequent evaluation determined that the root cause of the event was inadequate labeling of the detectors. Actions were taken to label the detector locations with the respective "North" and "South" orientations.

The inspector reviewed licensee event report (LER) 87-02, submitted by letter dated May 27, 1987, to verify the report accurately described the event and the licensee's corrective actions. No inadequacies were identified.

5.4 Inoperable Equipment

Actions taken by plant personnel during periods when equipment was inoperable were reviewed to verify: technical specification limits were met; alternate surveillance testing was completed satisfactorily; and, equipment return to service upon completion of repairs was proper. This review was completed for the following items: (1) "D" Service Water Pump - April 7, 1987; (2) "B" RHR Pump - April 20, 1987; (3) stack gas channel II - April 28, 1987; (4) "D" RHR Pump - May 4, 1987; (5) "A" RHR Pump - May 11, 1987; (6) "C" RHR Pump - May 18,1987; (7) "A" Main Steam Line Radiation Monitor - May 27, 1987; and, (8) "A" toxic gas monitor - May 29, 1987. No inadequacies were identified.

5.5 Worker Contamination

A reactor operator became slightly contaminated on April 15, 1987 at about 2:30 p.m. while removing system tags per Switching & Tagging Order 87-262. The operator was clearing tags and opening suction valves to return the reactor water cleanup pump "B" to service following maintenance per MR 878-338 to fix the pump seals. A threaded fitting leaked on the line to the seal package causing reactor water to spray on the operator. The operator immediately closed the suction valve to stop the leak. Several gallons of water were spilled on the floor. The operator was not injured, but did become slightly contaminated on the wrist and back of the head to levels of 1500 and 500 corrected counts per minute, respectively, using an RM-14 survey instrument. The affected areas were immediately decontaminated and a whole body count showed no intake of radioactive material. The "B" RCU pump was subsequently returned to service without incident. No inadequacies were identified.

Two maintenance workers became contaminated while performing grinding operations during the disassembly and repair of the "B" RHR pump at 11:30 a.m. on April 21, 1987. The inspector reviewed the event and reviewed the licensee's investigation and evaluation of the incident, as described in a memorandum by the ALARA engineer dated May 10, 1987.

The men were working under RWP 87-270 to clean the pump flange and inner wear ring. Work was done initially in respirators. Contamination levels on the work surface ranged from 50K dpm/100 cm-sq to 160K dpm/100 cm-sq. The HP technician removed the workers from respirators at 10:50 a.m. after getting the results from two breathing zone air samples that showed air concentrations to be about 1.2-1.9 X 10-9 uCi/cc, which was less than the established limits (3 X 10-9 uCi/cc). The men worked without respirators until about 11:30 a.m., and used various materials to clean the pump surfaces during the interim period. The subsequent work on the pump surfaces with stone and scotch brite pad cleaning tools apparently generated fine particle dust that increased airborne radioactivity levels after the respirators were removed.

Facial contamination was found on both men at 11:30 a.m. as they left the work area. Nasal smears showed 680 and 67 ccpm respectively using ludlum counters. One individual took a whole body count after showering and did not have any intake of radioactive material. The whole body count for the second individual showed a minor intake with a 2.4% MPOB for the lungs and a 4.5% MPOB for the GI tract. A total of 7.1 MPC hours was assigned to the individual as exposure. The estimated total dose to the GI tract was about 9.4 millirems over 50 years, using the methodology of ICRP-30.

The licensee evaluated the cause for the above exposure and took appropriate corrective actions. Respirators were used during subsequent cleaning work on all four RHR pumps. The inspector noted that the unintended exposure could have been avoided had respirators been used throughout the cleaning activity, or had a followup air sample been taken after changing the tools used to do the work. However, the inspector also noted that the exposure was very small in comparison to the allowable regulatory limits. No inadequacies were identified in the licensee's followup actions.

5.6 Release of Contaminated Tools from the RCA

The licensee notified the inspector on April 23, 1987 of the results of a radiation survey of designated "clean" tool storage areas outside the radiation controlled area (RCA) but within the protected area that identified 22 items with very slight amounts of fixed radioactivity on them. All tools having fixed contamination had readings between 150 and 400 ccpm using a RM-14 with and HP-210 probe, with the exception of two hammers that had levels of 15,000 and 12,000 ccpm. The corresponding radiation levels on the hammers were about 0.2-0.3 mR/hr gamma, and 2.5-3.0 mR/hr beta. The dose rate levels were contact readings and did not represent the potential for whole body dose rates at that magnitude. The licensee removed the items from the clean storage areas.

The inspector reviewed the results of the licensee's surveys and the planned actions to investigate the source of the material and how it became uncontrolled. The above materials were discovered during a final survey of the tools prior to release from the owner controlled area as they were being transferred to a contractor's shop offsite. The licensee completed an additional survey of the contractor's Brattleboro fabrication shop during the period from April 29-30, 1987 using an Eberline ESP-1 sodium iodide detector to scan the storage areas, and an RM-14 survey instrument to investigate in detail any items with activity identified from the general area survey. All items showed background levels on the RM-14, with the exception of two shackles, an air hose and two tube-loc knuckles which had activity between 50 to 80 ccpm fixed contamination using the RM-14. Although the radioactivity on the tools was less than the acceptable release limit of 100 ccpm on the RM-14, the inspector requested the licensee to return the material to the Vermont Yankee site pending completion of the review of the matter. The licensee returned the material to the site on May 4, 1987 and either decontaminated the materials completely or retained them on site for designated use within the RCA.

The licensee's followup investigation of this matter was described in a memorandum by the Plant Health Physicist dated May 12, 1987. Although it could not be concluded how or when each individual item with fixed contamination was removed from the RCA, the licensee concluded that the two hammers with elevated contamination levels were probably not frisked, and the remaining items were most likely frisked in an area with a relatively high background area. Immediate corrective actions taken were described in a May 12, 1987 memorandum to the Radiation Protection staff which assured that materials removed from the RCA are frisked in a low background area. The additional controls include a requirement that all items removed from the RCA that were in a contaminated area must be frisked by a HP technician. The inspector determined that the corrective actions would be effective to preclude inadvertent release of materials with fixed contamination in excess of the established limits. The inspector noted that the licensee identified additional steps in the May 12, 1987 memorandum that are under consideration to further control the release of material from the RCA. The licensee requested that a meeting be arranged with the NRC technical staff to discuss the proposed measures.

The inspector noted that, based on the magnitude of radioactivity identified on the tools, no regulatory limits were exceeded, and none of the items created a safety or health hazard to plant workers or members of the public.

The technical specifications define contamination as "fixed contamination shall be considered significant and unreleaseable from the owner controlled area if the dose rate on any accessible surface exceeds 0.5 mR/hr. Additionally, material is unreleaseable from the owner controlled area if fixed contamination exceeds 22,000 dpm per 100 cm-sq". However, these limits are higher than the accepted industry standards for release limits of 100 ccpm using the RM-14, which is the criteria used in licensee administrative procedure AP 0521 for the control of material released from the RCA. Based on the above, the inspector noted that although no regulatory limits were exceeded, the licensee's procedural limits in AP 0521 were not met when the tools found outside the RCA but within the protected area were released. Neither regulatory nor licensee administrative limits were exceeded for the tools found in the Brattleboro fabrication shop.

The failure to meet the requirements of AP 0521 constitutes a violation of Technical Specification 6.5.B that was identified by the licensee. This item is considered admissible as a licensee identified violation in that there is no health or safety significance, and since at least four of the five criteria in 10 CFR 2 are satisfied. However, the inspector questioned whether the item could reasonably have been prevented based on corrective actions taken for previous violations. Inspection item 87-04-03 documents the inspector's previous concerns in this area and describes the previous performance history. This item will continue to be tracked as Inspection Item 87-04-03, which remains unresolved pending further NRC review of the licensee's controls on a subsequent inspection by an NRC Health Physics inspector.

5.7 Review of Jumpers and Lifted Leads

Jumper and lifted lead (J/LL) requests 87-10 to 87-25, and 86-156 and 86-157 were reviewed to verify that controls established by AP 0020 were met, no conflict with the technical specifications were created, the requests were properly approved prior to installation, and a safety evaluation in accordance with 10 CFR 50.59 was prepared if required. Implementation of the requests was reviewed on a sampling basis. The matters related to J/LL 86-156 and 157 warranted further followup, as discussed below in section 7.0. No inadequacies were identified.

5.8 Review of Switching & Tagging Operations

The switching and tagging log was reviewed and tagging activities were inspected to verify plant equipment was controlled in accordance with the requirements of AP 0140, Vermont Local Control Switching Rules. Action completed under Orders 87-262, 87-320, 87-333, 87-304, 87-278, 87-264, 86-1027 and 87-364 were reviewed. No inadequacies were identified.

5.9 Review of Potential Reportable Occurrences

The inspector reviewed a PRO 87-17 dated April 16, 1987 regarding a potential inadequacy identified with pressure transmitter PT2-3-56C&D, which are used on the reactor vessel to sense reactor pressure and provide a low pressure permissive for actuation of the ECCS systems when reactor pressure reaches 350 psig. An engineering review of the installation identified that the transmitters tapped off the reactor vessel using the cold reference leg of the vessel level instrumentation system, which is at a higher elevation than the tap off point shown in engineering drawing G191267 and used for calibration of the instrument. The higher tap off point would cause the trip setpoint to be nonconservative due to the additional elevation head difference. The licensee determined that the corrected trip setpoints from the transmitters were 317 and 319 psig, which were within the range of 300 psig to 350 psig required by the technical specifications. Based on the above, the instruments were considered acceptable and the item was not reportable. No inadequacies were identified.

The inspector reviewed PRO 87-19 dated April 23, 1987 which concerned the potential inoperability of the electric fire water pump during the period between performance of the normal monthly surveillances in February and March 1987. The fire pump was initially evaluated to have been found inoperable on March 23, 1987 based on an auxiliary operator report that isolation valve FP-13 was in the closed position, thus isolating the pressure switch from the fire header and negating the automatic start feature of the pump on low header pressure. The licensee initially concluded that the item was reportable under 10 CFR 50.72(a)(2)(i)(B). However, subsequent licensee investigation of the event identified conflicting evidence regarding the operability of the pressure switch and the initial findings were discredited. The inspector reviewed the piping configuration, the sequence of events reconstructed by the licensee and the bases for the licensee's determinations. No inadequacies were identified.

5.10 Recirculation Pump Trip

The "A" recirculation pump motor generator (MG) set tripped off line at 9:18 p.m. on May 24, 1987 while the reactor was operating at 98% full power in end of cycle coastdown. The MG set tripped because of a low lube oil pressure condition. Plant operators followed operational transient procedure OT 3118 to stabilize the reactor using the operating "B" recirculation pump at about 50% power. Following actions to correct the lube oil problem, plant operators restarted the "A" recirculation pump using procedure OP 2110 at 10:28 p.m. on May 24, 1987 and power operations continued. The inspector reviewed the plant response to the transient using the control room strip chart recorders and the operators actions for conformance with the operating procedures. No inadequacies were identified.

The licensee determined that the low lube oil pressure condition occurred due to the closure of manual isolation valve LO-7A on the suction side of "C" lube oil recirculation pump for the "A" MG set. The manual valve "drifted" closed slowly over a period of time because of small amplitude but high frequency vibrations inherent in the piping that caused the hand wheel to oscillate and thereby "hammer" the valve closed slowly over a long period of time. The licensee took actions to secure the handwheel and valve in the open position, and to verify proper position of all lube oil valves. Subsequent actions were also taken to tighten the packing on the lube oil pump suction valve.

The inspector reviewed the lube oil piping and valves and independently verified the licensee's conclusions for the cause of the recirculation system pump trip were correct. The inspector also noted that the standby "A" lube oil pump would not have automatically started in response to the trip of the "C" pump due to the design of the control circuitry. No inadequacies were identified.

5.11 Failed Fuel Indications

The inspector reviewed control room panel recorders and offgas sample analysis results to monitor the status of offgas radiation levels and release rates during the inspection period. Offgas release rates remained within the range of 8000 to about 10,000 micro-Ci/sec, which were well below the Technical Specification 3.8.K.1 limit of 0.16 Ci/sec. No increases were noted in the stack gas monitor readouts.

The inspector noted based on discussions with reactor engineering personnel that fuel sipping operations are planned during the 1987 outage to locate the defective bundle(s). The licensee's initial plans are to sip 152 bundles from core reload batches having fuel burnup of about 20,000 MWd/MTU. The licensee estimated from the present offgas rates and slopes that two or three fuel pins are defective. The results of the licensee's fuel sipping operations will be examined during the routine inspection program. No inadequacies were identified.

5.12 TBCCW System Tritium Contamination

The inspector reviewed licensee actions to investigate the source of tritium in the turbine building closed cooling water (TBCCW) system, which contains tritium at levels of about 1.0 X 10-4 uCi/ml. Measurable tritium levels were detected in the system in September, 1986. Several flushes of the system since 1986 have not reduced the levels. The source of the leak has not been identified, although the reactor feedwater pump coolers will be investigated during the outage shutdown for possible leaks. The inspector noted that, based on the amount of makeup to the TBCCW system, it has been essentially leak tight since the system contamination first occurred, and any water flushed from the system is processed by the licensee in the radioactive waste systems on site. Based on the above, there has been no loss of control of tritiated water.

The inspector had no further comments on the licensee's actions at the present time. This matter will be followed on subsequent routine inspections.

6.0 Storage Pit Wall Weepage

The status of licensee actions on this item was reviewed with operations personnel on April 10, 1987. The leakage source was identified to be from demineralized water piping embedded in the concrete on the West wall of the dryer-separator storage pit, which feed service boxes along the side of the pit. Actions were taken to isolate the West header isolation valve using a blocking tag for the operations shift supervisor. A maintenance request was issued to either repair the leak or permanently isolate the header. Licensee actions to isolate the leakage were acceptable.

The licensee reviewed the other potential leakage paths for water from the demineralized water header to determine where else leakage may have drained, and specifically, whether water may have migrated between the sand bed and the drywell liner. The licensee determined that no leakage migrated between the liner and the sand bed, based on a check of the drain piping in the areas of interest that was found to be dry. Additional actions are planned by maintenance personnel to complete boroscope examination of the drain lines to assure they are not plugged. The inspector noted that this action will also be taken based on a commitment made to the NRC staff in response to Generic Letter 87-05, as described in letter FVY 87-52 dated May 8, 1987. The inspector noted that the commitment to complete the inspections was covered by the licensee's commitment tracking system, based on a memorandum dated May 26, 1987. No inadequacies were identified.

7.0 Leak Rate Testing of Torus Penetrations

The licensee informed the inspector during this inspection of his intentions to change out torus thermocouples during the 1987 outage to install environmentally qualified units to address the requirements of Regulatory Guide 1.97 and to address a discrepancy described in Nonconformance Report 86-114 dated 12/18/86. The licensee stated that replacement of the thermocouples in the torus penetrations would open the containment pressure boundary but that no Appendix J type leak rate was planned or considered necessary following the changeout due to the nature of the penetration geometry and thermocouple fitting. The licensee stated that the thermocouple penetration geometry used a 3/4 inch NPT type tapered thread fitting that was similar to the test fittings the NRC exempted from Appendix J leak rate testing, as documented in the NRC's August 19, 1987 letter and SER granting certain exemptions from the requirements of 10 CFR 50.54(o). The inspector's followup review of this item noted the following.

While investigating requirements for a planned design change in December 1986, the licensee determined that existing (installed) thermocouples TE-16-19-30 and TE-16-19-34 were different than the original plant equipment, and that the environmental qualification of two torus thermocouples was indeterminate. Of the two, only TE-16-19-30 penetrates containment boundary and presented a containment boundary concern. The original plant equipment was non-spring loaded thermocouple elements designed for use without thermowells. The existing units installed in 1981 per MR 81-727 were spring loaded units designed for use in thermowells. The installed configuration of the existing units made the environmental qualification of the thermocouples indeterminate, and raised questions regarding the adequacy of the penetration for containment boundary purposes and the operability of the thermocouples to satisfy technical specification Table 3.2.6 requirements for post accident instrumentation. This discrepancy was documented and addressed in Nonconformance Report 86-114.

The licensee's engineering organization, in conjunction with Yankee NSD, evaluated the discrepancy and determined that the existing thermocouples were acceptable pending replacement at the next refueling outage. The basis for the conclusion was provided in a Justification for Continued Operation attached to NCR 86-114. For the type of thermocouple installed in the torus, the thermocouple head is the primary containment boundary. The sealed thermocouple head was considered an acceptable containment boundary because it was qualified and pressure tested by the vendor to 100 psi, and since the installed units have successfully passed two Appendix J Type A leak rate tests at 45 psig since installation.

J/LL Requests 86-156 and 157 were issued to to address the post accident instrumentation issue by providing inputs from other known environmentally qualified thermocouples to control room recorder TR-16-19-45 for readouts of drywell air temperature and torus air temperature. Specifically, for drywell air temperature, the existing input from TE- 16-19-30 was replaced by the input from TE-149-1, which measured drywell return air temperature to RRU#1 and is indicative of overall drywell air temperature. Additionally, for torus air temperature, TE-16-19-34 was replaced by the input from TE-16-19-41, which provided torus air temperature to an indicator on CRP 9-3. The CRP 9-3 indication will have to be fully environmentally qualified per Regulatory Guide (RG) 1.97 requirements during the 1987 refueling outage.

The inspector reviewed the licensee's safety evaluation and justification for continued operations and identified no inadequacies. Full compliance with the RG 1.97 requirements is scheduled for 1987. The inspector will follow the licensee's actions to install environmentally qualified thermocouples and fully meet the RG 1.97 instrumentation requirements as part of the NRC review of the 1987 outage design changes. This item is unresolved pending completion of license actions to meet RG 1.97 requirements and subsequent review by the NRC (UNR 87-09-01).

The inspector noted the licensee's position regarding post modification containment leak rate testing of the torus penetrations. However, 10 CFR 50, Appendix J Section IV requires that modifications of components that are a part of the primary reactor containment boundary be followed by a Type A, B, or C leak rate test as applicable for the affected area. Additionally, the inspector reviewed the design details for the torus penetration of interest (X215) and noted the similarity to leak rate test connections. However, the inspector questioned whether the configuration was of the type addressed and exempted by the staff's SER supplied with the August 19, 1983 letter, and specifically, whether it meets the requirements of a Type 4 exemption for instrument lines 1 inch or less in diameter meeting the requirements of RG 1.11. This item is unresolved pending further NRC review to determine whether an exemption from the requirements of 10 CFR 50 Appendix J is required for the licensee to implement the modifications as planned (UNR 87-09-02).

8.0 Fuel Handling Activities

The inspector reviewed fuel handling activities in progress in the spent fuel pool April 13-15, 1987 to empty out PAR racks #15 and #18. The racks were being emptied in preparation for movement to the Northeast corner of the spent fuel pool, to make room in the Southeast end of the pool for the insertion of one NES rack (see Inspection Item 87-06-02 and section 13.0 below for a discussion of the NRC staff review of that design change). The inspector verified that fuel handling activities were performed in accordance with procedures OP 1490, 1410, and 4102, and that the applicable requirements of Technical Specification 3.12 were satisfied. The inspector noted that communications, staffing and the use of tag boards was adequate, and that plant systems were aligned as necessary to support the activity. No inadequacies were identified, except as noted below.

The inspector reviewed the periodic checks performed on the refueling equipment and noted that none of the refueling interlocks applicable to fuel movement in the spent fuel pool had been completed, which included a functional verification of the grapple "closed" and "full up" interlocks, and a test of the main hoist overload interlock. The inspector noted that the grapple closed and full up functions were functionally tested on an on-going basis during fuel movement and were operable.

This matter was discussed with the Operations Supervisor on April 15, 1987. The inspector stated that the aforementioned checks could be viewed as required by OP 1410, Prerequisite Step 5, which references the refueling interlock tests required by OP 4102. The licensee stated that the functional checks of the refueling interlocks were not considered applicable for the present plant conditions, nor required by Technical Specification 3.12 and 4.12. The inspector acknowledged that none of the refueling interlocks designed to assure criticality controls were applicable for the plant conditions. The inspector further noted that the technical specification requirements do not clearly state which checks are required for fuel handling in the spent fuel pool with the reactor operating at power. The inspector maintained, however, that the grapple and hoist checks were important and applicable for fuel handling in the spent fuel pool.

The inspector stated that licensee procedures should be reviewed and changed as necessary to clearly identify the procedures, controls and prerequisite checks that should be completed for fuel handling activities in the spent fuel pool with the reactor operating at power. This action is particularly necessary in view of the extensive number of fuel moves (3400 plus) that will be required to rerack the spent fuel pool during power operations. The licensee acknowledged the inspector's comments and stated that the procedures would be reviewed and changed as necessary prior to the fuel movements to rerack the pool. This item is unresolved pending completion of the licensee's actions and subsequent review by the NRC (UNR 87-09-03).

9.0 Semiannual Effluent Release Report

The Senior Chemistry and Health Physics Engineer notified the inspector on April 10, 1987 of a problem that was identified in meeting the Technical Specification 6.7.C.1 reporting requirements for the semiannual effluent release report for the period of July-December, 1986. The initial report filed on March 1, 1987 did not provide calculated doses from plant effluents or plant meteorological data for the subject reporting period. The licensee had intended to provide the subject information in a supplemental report as allowed by the technical specifications. However, a review by Yankee NSD on April 8, 1987 determined that the specifications apparently do not contain a provision that allows reporting of meteorological data in a supplemental report, only the calculated dose rate data.

The inspector noted that the supplemental Semiannual Effluent Release Report was issued by letter FVY 87-58 on May 26, 1987, and that the report provided information on meteorology as well as the summary of estimates of the offsite doses. The inspector reviewed the PRO for this item dated April 14, 1987. The licensee concluded that the item was not reportable. The inspector noted that the specifications do not prohibit reporting of both meteorological data in the supplemental report along with the dose estimates. Additionally, the subject information was generated, on file with the licensee, and therefore available for NRC review upon request. No inadequacies were identified.

10.0 Surveillance Testing

The inspector reviewed surveillance test results completed during the inspection period to verify testing was performed by qualified personnel in accordance with established procedures, and that test results demonstrated components and systems under test were operable. The following test results listed below were reviewed. The RHR test results were reviewed for each pump ("A" through "D") as the operability demonstration on the pumps following maintenance to replace the pump impellers.

-- OP 4124.04, RHR Pump Operability Data Sheet

-- OP 0202.01, RHR Pump Vibration Measurement
-- OP 4124.10, RHR Pump Vibration Data Sheet
-- OP 4124.1.02, RHR Pump Operability Demonstration
-- OP 4511.18, Source Check of AOG Monitors 3127 and 3128, 5/12/87

The following item warranted additional review.

The results of post maintenance testing of the "B" RHR pump on April 25, 1987 showed high vibration on the motor upper bearing at 4000 gpm system flow. The vibration had displacement and velocity values of 8 mils and at 0.35 inch/sec, respectively. The vibration occurred at about a frequency of 15 hz. The motor vibrations were acceptable at higher flow conditions. The RHR system is normally operated in the flow range of 6500-7200 gpm, but could be operated at lower flows under certain conditions when the system is used in the containment cooling mode. The licensee's review of this matter was summarized in a memorandum to the Maintenance Supervisor dated April 29, 1987. The licensee investigated the cause of the condition and concluded that the pump and motor were in good mechanical condition, and that the high vibration was due to the excitation of a pump resonance. The licensee determined that all four pumps have a natural resonance at close to one half the operating frequency of the motor. The licensee concluded that the vibrations would not adversely affect pump operation and that the pump was capable of performing its intended safety functions.

The inspector independently reviewed the pump operation and vibration frequency spectrum data and identified no inadequacies.

11.0 Maintenance Activities

Maintenance activity was reviewed to determine the scope and nature of work done on safety related equipment and equipment referenced in the technical specifications. The review confirmed: the repair of safety related equipment received priority attention; technical specification limiting conditions for operation were met while components were out of service; performance of alternate safety related systems was not impaired; and, the activity was completed in accordance with established procedures and plant equipment controls.

Maintenance activity associated with the following was reviewed to verify (where applicable) procedure compliance and equipment return to service, including operability testing.

-- MR 87-651, Condensate Storage Tank Inspection
-- MR 87-615, "D" Service Water Pump Impeller Replacement
-- MR 87-809, "A" Service Water Pump Impeller Replacement
-- OP 5200.33, RHR Pump Disassembly and Reassembly
-- OP 5200.34, RHR Motor Disassembly and Reassembly

The items discussed below required further followup.

11.1 Condensate Storage Tank Inspection, MR 87-651

This maintenance work involved the underwater ultrasonic inspection of the condensate storage tank (CST) floor to determine whether additional corrosion had resulted in further degradation of the tank bottom. The inspection results showed that the thickness of the aluminum plates in the tank floor remained within the nominal dimension of 0.250 and 0.340 inches for the center and periphery floor plates, respectively. Based on these inspection results, the licensee determined it is not necessary to repair the CST floor plate during the 1987 refueling outage.

The inspector reviewed the safety evaluation completed by the licensee to perform the underwater inspections at a time when the CST remained operable as a suction path and water source for the RCIC, HPCI and core spray systems. The licensee implemented controls to protect the divers and to assure the plant safety systems remained operable. The inspector verified loose parts were controlled when admitted into the CST, and that tools and materials were removed upon completion of the inspections. No inadequacies were identified.

11.2 Service Water Pump Impeller Replacement

The licensee completed actions during the inspection period to replace the impellers on the "A" and "D" service water pumps after determining that impellers having an incorrect diameter were installed during recent maintenance work to overhaul the pumps. The licensee's investigation into the circumstances were documented in a memorandum by the Maintenance Senior Engineer dated March 13, 1987.

The licensee determined that replacement impellers received under Purchase Order 26269 were oversized at 11-15/16 inches in diameter, when the proper replacement size was 11-5/16 inches. The licensee determined that the wrong sized impellers were purchased due to an error made by maintenance personnel by specifying the wrong dimensions when filing a replacement order.

Installation of the oversized impellers resulted in operation of the pumps at a hydraulic point different from the pump performance curve provided by the manufacturer, and resulted in greater flows from the pumps. The extra flow caused the pump motor to expend additional work, which resulted in higher full load amperage through the motor windings and consequent higher winding operating temperatures. Motor winding temperatures remained at or below the limits set by the NEMA ratings for the winding insulation.

The licensee's concluded that the service water pumps remained operable during periods of plant operation with oversized impellers installed. The licensee replaced the oversized impellers to avoid excessive erosion in the pumps, which could cause a maintenance problem over long term operations. The licensee also reviewed the purchase order file for the service water pumps and verified that the proper sized impellers were received on previous purchase orders, and that material in stock are the proper replacement parts.

The inspector reviewed the licensee's evaluations and corrective actions, and identified no inadequacies.

11.3 RHR Pump Wear Ring Inspection and Repair

The licensee completed actions during the period from April 20 - May 22, 1987 to inspect and overhaul the RHR pumps, and to replace the RHR pump impellers in the sequence of B, D, A and C, respectively. The inspections were completed to satisfy a commitment to the NRC to determine whether IGSCC cracking of the pump wear rings existed at Vermont Yankee. The cracking problem had been observed at other facilities in the Bingham-Willamette Model CVIC 16X18X26 vertically mounted, centrifugal pumps. The inspector reviewed the maintenance activities to verify the requirements of OP 5200.33, and the licensee's work control procedures were followed, as well as the conditions specified in the NRC safety evaluation provided in a letter dated March 12, 1987 granting relief from the requirements of Technical Specification 4.7.A.3. No inadequacies were identified in the work control process or in the actions taken to satisfy NRC commitments.

The following is a summary of the major inspection findings for all four RHR pumps, as obtained from VT-1 Visual Examination Reports dated April 20, May 5, May 15 and May 19, 1987.

Pump	Outage	Major Findings
"B"	April 20-30	+ no visible wear ring cracks + 5 inch through wall impeller crack + suction recirculation erosion evident
"D"	May 4-9	+ no visible wear ring cracks + 5 inch crack in impeller + suction recirculation erosion evident
"A"	May 11-15	+ visible cracks in 1 of 4 wear rings + no impeller cracks + suction recirculation erosion evident
"C"	May 18-22	+ no impeller cracks + no visible wear ring cracks + suction recirculation erosion evident

Cracking was identified in the lower stationary wear ring (one of four) on the "A" RHR pump. Radial cracks were identified on the outer side of two of eight bolt holes; radial and a crescent shaped cracks were identified on the outer side of a third bolt hole. The licensee determined that cracks of the type observed at other facilities were not evident from the visual inspections of the wear rings from any of the pumps. The wear ring with cracks on the "A" RHR pump was a stationary ring, cast from a material not susceptible to IGSCC cracking. The inspector noted the licensee's inspection results and conclusions, but stated to the licensee that microscopic examinations should be completed on the wear rings to determine whether cracks are present at the incipient stage.

The pump impellers were made per ASTM A296 (presently ASTM A743) specifications and are Grade CA15 martensitic 400 series cast stainless steel. The crack in the "B" pump impeller was circumferential, about 5 inches long, and penetrated through wall in the 3/8th inch thick shroud. The crack appears to follow a weld in the shroud, which appears to have been a weld repair of an original casting flaw. The 5-1/4 inch crack in the "D" impeller was not through wall, and was located on the lower side of the hub shroud. Neither crack crossed a flow vane. The licensee concluded that the cracks were most likely defects in the original castings that went undetected following manufacture in the late 1960's and became visible (uncovered) from surface erosion during the 15 year plant operating period.

The licensee evaluated the inspection findings for the "B" and "D" pumps as of May 5, 1987, and determined based on input from the pump manufacturer and Yankee NSD, that the flaws would not have adversely affected the operability of the pumps in the as-found conditions. The bases for the licensee's conclusion was contained in an engineering evaluation and justification for continued operation (JCO) dated May 5, 1987, which was appended to a Potential Report Form for May 7, 1987. The inspector attended a meeting of the Plant Operations Review Committee on May 5, 1987, during which the plant staff addressed the engineering JCO. The bases for the licensee position was reviewed by the inspector during the inspection period, and by members of the NRC Region I and NRR staffs during telephone conference calls on May 8 and June 1, 1987. The NRC staff concurred in the licensee's conclusions based on: the flaws appear to be casting defects; the relative size and location of the flaws in the impeller assembly; the general toughness of the 410 series cast stainless steel materials; the good performance experienced by the pumps as demonstrated by the vibration monitoring program; and, the good service history reported by the vendor for impellers with flaws of the type observed. The final engineering evaluation and the need for subsequent NRC staff action will be determined by the results of the destructive examination planned for the defects. No inadequacies were identified.

The inspector reviewed the purchase order package for PO 28567 to review the manufacturing history for the four new impellers and to verify that material test reports and vendor certifications were provided in accordance with the requirements stipulated by the licensee. This review also verified that the replacement parts were manufactured with a 215-265 BHN, which has been determined to be less susceptible to IGSCC cracking. The inspector noted that the impeller work package documented a history of producing unacceptable castings, but that process controls were applied to assure acceptable quality in the final product. Casting quality was assured by both visual inspections and by magnetic particle inspection of the final castings. Weld repair of the new impellers was only for minor surface casting flaws. The inspector conducted a visual examination of the impellers installed in the "A" and "C" RHR pumps during this inspection period, and noted the components were of acceptable quality with hydraulically smooth surfaces throughout. There were no casting defects of the type observed on the old impellers from the "B" and "D" pumps. No inadequacies were identified.

The inspector noted that the licensee plans to conduct destructive examination of the defects in the components removed from the RHR pumps, but the scope and nature of the inspection plans are still being developed. The NRC staff requested the licensee to provide a component with flaws for independent evaluation of the defects at Brookhaven National Laboratories. A collaborative effort is under consideration. The inspector noted that the examinations should also include the wear rings. The inspector further noted that, based on the recirculation flow erosion effects observed on the suction side of several impellers, the licensee should re-evaluate the previous engineering conclusions regarding the adequacy of the RHR minimum flow lines in light of the present inspection results (reference inspection item 86-25-02).

This item is unresolved pending completion of the licensee's destructive examination and/or evaluation of the RHR pump defects to verify no abnormal component degradation has occurred due to service induced conditions, and subsequent review by the NRC (UNR 87-09-04).

12.0 Residual Heat Removal (RHR) Pump Inspection/Repair Preparations

During this inspection period the inspector (region-based project engineer) reviewed the licensee's enhanced monitoring program, plans and procedures for inspection and repair of the four RHR pumps. The RHR pump enhanced monitoring program was developed in response to both the licensee's and the NRC staff's identification of a potential generic problem with wear rings for Bingham-Willamette Model 16X18X24 CVIC pumps. Specific details of the technical concerns and regulatory considerations regarding the licensee's monitoring program have been described in NRC Region I Inspection Reports, commencing with 85-40 (as Inspection Item 85-40-09) and most recently updated in Inspection Report 87-06. As discussed in the latter report, the licensee had requested and was granted relief, subject to certain specified conditions, from the containment leakage testing requirements of Technical Specification 4.7.A.3, in order to perform the RHR pump wear ring inspection and replacement in a safe and efficient manner.

The inspector reviewed the licensee's procedures that were written to control and perform the RHR pump and motor maintenance activities, as listed below:

-- Operating Procedure (OP) 4124.1, Revision 1, "Preparation and Testing of RHR Pump for Wear Ring Inspection/Repair"

-- OP 5200.33, Revision 1, "RHR Pump Disassembly and Reassembly"

-- OP 5200.34, Original, "RHR Motor Disassembly and Reassembly"

Additionally, the inspector reviewed applicable referenced vendor technical manuals and drawings, and discussed the specific procedural controls and steps with licensed operators and the responsible operations and maintenance engineers. Based on this review, the inspector verified that the NRC staff's conditions (as delineated in Inspection Report 87-06) were appropriately incorporated in the procedures. The procedures were well written and reflected thoughtful consideration of the steps necessary to assure proper control and performance of the wear ring inspection/repair of each RHR pump, inspection and maintenance of each RHR motor, and subsequent testing activities.

Licensee personnel who would be controlling or performing the planned maintenance activities were very knowledgeable of all aspects of the RHR pump/motor inspections, including provisions for assuring that system boundary integrity was properly controlled, that leakage was monitored, and that contingency actions would be implemented should measurable leakage be detected. The inspector noted that no boundary leakage occurred during the work on each of the four RHR pumps.

The actual RHR pump inspection and maintenance activities were inspected, in part, by another NRC region-based inspector (as described in Inspection Report 87-08), and by the resident inspector, as described in section 11.0 above. No violations or safety concerns were identified during the review of the procedures and the preparations for the work.

13.0 Spent Fuel Pool (SFP) Changes

The inspector continued a review of licensee plans and procedures to install one NES rack in the SFP per EDCR 87-405 (reference: Inspection Item 87-06-07). The inspector noted during this inspection that the licensee will use the old lifting device to relocate the existing PAR racks #15 & #18 within the pool to make room for the insertion of the NES rack. Since the old lifting fixture is not designed to meet the criteria of NUREG-0612, the licensee will limit the lift height of the racks in the pool to less than 4-1/2 inches to assure stress limits on the pool floor will not be exceeded in the event of a drop of the rack during the transfer. The inspector reviewed the Installation & Test (I&T) Procedure for EDCR 87-405 to verify controls for a safe lift identified in the PAR Rack Drop Analysis were incorporated.

The inspector noted that stipulations were included to limit lift height to 4-1/2 inches and to use a choker of 3 foot minimum length to limit the interaction between the rack and the lifting fixture. However, the I&T procedure did not include a precaution for rigging personnel to keep the load vertical at all times. The inspector discussed this item with a licensee representative and noted that maintaining a vertical posture during lift would be assured by the symmetry in the design of the lifting fixture and due to general practice by rigging personnel. The licensee stated that a note would be added to the I&T procedure to explicitly caution rigging personnel to keep vertical posture on the load.

The inspector noted further that the licensee's design evaluation and safety analysis demonstrated a satisfactory criticality analysis for both the PAR and the NES racks using the present fuel types with an average enrichment of 2.89 weight percent (wt %) U235. The criticality analysis showed the Keff for the racks was less than the NRC staff acceptance criteria of 0.95. The criticality analysis was performed for the PAR racks assuming 3.0 wt % U235. The criticality analysis was performed for the NES racks assuming 3.25 wt % U235. The inspector noted that the rack analysis assumptions were bounding for the present fuel design, and for the transition fuel bundles with 2.99 wt % U235 that will be received on site in June, 1987 and used for cycle 13 operation beginning in the Fall of 1987. However, the licensee plans to use high energy bundles with 3.24/3.26 wt % U235, that will be received on site in the Fall of 1988 and inserted in the reactor in the Spring of 1989. The licensee has noted that, since the PAR racks were analyzed with 3.0 wt % fuel, the simultaneous use of PAR racks to store the high energy bundle design should be prohibited until the additional analysis is performed.

The item is open pending further completion of licensee actions to perform the additional criticality analysis for the PAR racks, and further NRC staff review of the controls to keep the high energy bundles out of the PAR racks pending completion of the analysis. This is considered the first part of Inspection Item 87-09-05.

The inspector noted further that the licensee had addressed the acceptability of the present Technical Specification 5.5.E limit for the present and transition fuel design. The inspector reviewed the licensee's calculations that demonstrated that fuel with 2.89 wt % or less enrichment will not exceed the present limit of 16 grams per longitudinal centimeter of the fuel assembly. Additionally, assuming fuel design parameters of 1.044 cm pellet diameter, 3.0 wt % average enrichment and 10.48 gm/cm-cu pellet stack density, the inspector verified that the limit of 16 gm/cm will not be exceeded. However, the licensee concurred that the TS 3.5.E limit may have to be changed to store fuel in the pool having 3.24/3.26 wt %, which could have local enrichments as high as 3.46 wt % in axial slices of the fuel rods. The licensee stated this matter would be addressed in a future licensing action to address the use of high energy fuel bundles in the core, and that consideration was being given to replacing the limit on U235 loading in the fuel with a limit on the K-infinite for the bundles.

The inspector identified no inadequacies with the licensee's plans or calculations. However, this item is considered open pending the completion of the licensee actions to address the adequacy of the present Technical Specification 5.5.E limits for storing the high energy fuel bundles in the spent fuel pool.

Licensee actions to address PAR rack criticality analyses and changes to license limits for fuel bundle U235 loading required as a result of the high energy fuel design is considered an unresolved item (UNR 87-09-05).

14.0 Management Meetings

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