U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 86-01

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

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

Facility Name: Vermont Yankee Nuclear Power Station

Inspection At: Vernon, Vermont

Inspection Conducted: January 2 - February 3,1986

Inspectors: William J. Raymond, Senior Resident Inspector
Thomas B. Silko, Resident Inspector

Approved by: L. E. Tripp 2/21/86
L. E. Tripp, Chief, Reactor Projects Section 3A Date

Summary: Inspection on January 2 - February 3, 1986 (Report No. 50-271/86-01)

Areas Inspected: Routine, unannounced inspection on day time and backshifts by the resident inspectors of: actions on previous inspection findings; physical security; shutdown operations; followup of outage activities; outage maintenance and testing; support modifications as part of the seismic reanalysis program; and, operating procedures for degraded grid voltage conditions. The inspection involved 202 hours by two resident inspectors.

Results: No violations were identified in 7 areas inspected. Reviews of outage operational activities identified no conditions adverse to safety. Further licensee action or followup is required to evaluate the environmental qualification of the containment hydrogen/oxygen analyzers; to assure outage contractors meet QA requirements; to assure the entire containment boundary is tested per 10 CFR 50 Appendix J requirements; to assure the reset feature of the RPS is operable prior to core reload; and, to assure degraded grid alarm response procedures are issued prior to plant startup.

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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.

Vermont Yankee

Mr. P. Donnelly, Maintenance Superintendent
Mr. J. Pelletier, Plant Manager
Mr. R. Wanczyk, Technical Services Superintendent

Morrison & Knudsen

Mr. C. Chen, Welding Engineer
Mr. J. Harriston, QA Manager

Meetings were held with the Vermont State Nuclear Engineer on January 9 and 24, 1986 in the NRC Resident Office to discuss NRC inspection of outage activities and recent events. The following items were also discussed: the status of licensee actions and NRC review of the embedded baseplate and recirculation pipe whip restraint problems; weld wire issue controls and control of welding repairs; the recent weld production problems on the N2 nozzles; the scope and schedule of NDE inspections by the NRC van that was onsite for the period of January 21-31, 1986; and, the NRC actions regarding written notifications to Vermont State following enforcement conferences with the licensee.

A joint tour was conducted of the torus room to observe installation details of the failed RSW-H164E support on the service water system.

The meeting and tour were beneficial for the review of items of mutual interest.

2. Summary of Facility Activities

The plant remained in a shutdown condition for the recirculation pipe replacement outage. The most significant licensee activity during the period was the continued installation and welding of the new recirculation system. NRC inspection of this work continued during the period and included a review of completed welds by a team of NRC Region I inspectors with the NDE van facilities. The NDE inspection included 18 radiographic examinations, 19 visual examinations, and 12 liquid penetrant examinations of field and Hatachi shop welds, and, 6 spool piece thickness measurements. The inspection results, documented in Report 86-02, identified no inadequacies in the licensee's NDE program.

The licensee stopped production welding on the recirculation N2 nozzle safe-ends on January 15, 1986 pending further evaluation of the reasons for welding problems attributable to apparent purge dam problems. Additional extensive mockup training on N2 nozzle welds was completed and an action plan to improve welding techniques was developed. The weld production schedule was revised to work large diameter pipe welds and thereby maintain critical path work activity. Following changes to address purge dam problems and weld technique (double beads on hot passes and changes in the torch angle), the licensee recommenced production welding on the N2 nozzles on January 31, 1986. The welding problems were also reviewed by regional inspectors, as documented in Report 86-02.

The licensee announced on January 24, 1986 a revised completion schedule for the current recirculation pipe replacement outage. The plant startup date was set back 55 days until June 26, 1986. The major reason for the slip in schedule was the delays in the installation of the new recirculation piping, with the problems in performing satisfactory welds on the N2 nozzles having the largest impact on production.

3.0 Status of Previous Inspection Findings

3.1 (Open) Violation 84-08-06: Inoperable HPCI System - reference Report 84-21. The inspector reviewed alarm response procedure OP 3140 and noted that the required operator actions for annunciator window CRP 9-3 A1 D-8 had been changed to require the HPCI logic reset pushbuttons be depressed following vessel level recovery below the high level trip setpoint. Completion of this change by the licensee satisfies this concern.

This item remains open pending completion of licensee actions to resolve concerns associated with the HPCI reset design feature, as discussed further in paragraph 8.2 below.

3.2 (Open) Follow Item 84-08-03: Station Batteries. This item was last reviewed in Inspection Report 85-30. The licensee has determined that the C&D Station Batteries are not susceptible to the same failure mechanism as Exide batteries during a seismic event. The inspector observed the C&D type LC-31 and LC-33 cell construction on January 21, 1986 with a YNSD Engineer, and noted that the internal seismic supports are provided that would hold the plates up from the bottom. The inspector noted further that IE Information Notice 83-11 dated March 14, 1983 addressed potential concerns with seismic vulnerability with old lead storage batteries. The inspector had no further questions regarding seismic concerns.

During the battery inspection on Janaury 21, 1986, the inspector noted that corrosion on Cell #16 of the "A" station battery had increased to the point that the top plate had cracked. The licensee reviewed the as-found conditions and concluded that the cell and bank were operable.

However, the expected performance of the cell under design load conditions was not addressed. The inspector noted that both station battery banks will be replaced starting in January, 1986.

This item remains open pending completion of licensee actions to replace the station batteries, and pending NRC review of outstanding battery issues.

3.3 (Open) Unresolved Item 85-25-06: Containment Electrical Penetrations. This item was last reviewed in Inspection Report 85-40. The licensee concluded that the as-found conditions in the containment electrical penetrations constituted a potential defect as defined by 10 CFR 21.21 and a report was filed on January 10, 1986. This item remains open pending completion of corrective actions by the licensee and subsequent review by the NRC, as discussed further in Paragraph 8.1 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; verification of 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 searches, and vehicle searches. No inadequacies were identified. The following item warranted further inspector followup.

4.1 Probable Tampering Event

The licensee notified the inspector on January 10, 1986 of his suspicions regarding the inadvertent actuation of fire protection deluge system for the auxiliary transformer at 2:38 P.M. on January 9, 1986. Plant operators responded to the actuation, noted that the deluge had been initiated manually, and secured the system since there was no fire. The deluge system is designed to spray a curtain of water on the transformer in the event of a transformer fire. Station power was provided by the startup transformers at the time and no damage to the plant resulted from the incident.

A 130 VDC ground (relative to station ground) on the B station battery occurred following the deluge actuation. The ground was cleared by plant operators by 3:00 P.M. on January 10, 1986 by opening breaker #9 on DC-2D after tracing the ground to the fire protection control circuit for the transformer. The ground was caused by moisture in the manual pull box used to actuate the deluge system and cleared after several days when the circuit dried out.

The deluge system was initiated from a manual actuation box mounted near the transformer on the outside West wall of the Turbine Building. After reviewing the event, the licensee concluded that the deluge system was manually actuated as an intentional act of vandalism or harrassment. The inspector attended a management briefing with plant security for this event at 4:00 P.M. on January 10, 1986 and followed the licensee response actions.

Plant security began an investigation starting on the morning of January 10, 1986 and interviewed personnel who responded to the incident and who may have frequented the area on January 9, 1986. Additional security patrols were established throughout the plant and remained in effect for some time after the event. The handling of personnel affected by a job action (see below) was reviewed to assure the individuals were processed offsite in a manner consistent with security requirements. Vermont State Police responded to a licensee request for assistance to obtain fingerprints from the manual pull box.

It was notable that the licensee's primary contractor for the recirculation pipe replacement outage had taken a job action against 49 welders within the 24 hour period prior to the incident. The job actions consisted of terminating 23 workers and suspending 26 others for three days. The workers who were suspended returned to the site on January 13, 1985, and some of the workers who were terminated also were rehired at a later date. The job action occurred following a disagreement over whether welders or fitters would be responsible for repairing cold laps in completed welds. However, no direct connection between the tampering event and the job action was ever established.

No other suspicious events occurred after the January 10th incident. The person responsible for actuating the deluge system was not identified. A security event report will be filed by the licensee. No inadequacies were identified in the licensee's response to the tampering event.

5.0 Review of Outage Activities

Plant tours were conducted routinely during the inspection period to review activities in progress and to verify compliance with regulatory and administrative requirements. Plant areas toured included the reactor building, the drywell, and the grounds within the protected area. The inspector attended outage meetings to keep informed of outage activities and the licensee's review and resolution of outage problems. Plant activities and events that warranted further NRC review are summarized below.

5.1 Shift logs and records were reviewed to determine the status of plant conditions and changes in operational status. During a review of control room logs, the inspector noted that on December 3, 1985, the diesel fire pump was tagged out of service for several hours while divers completed an inspection of the intake bay. The removal of the pump from service was noted in the Shift Supervisor's log, but not in the LCO tracking section of the shift turnover log. This matter was discussed with the shift supervisor on shift at the time the pump was tagged, who stated that an entry was not made in the turnover log since the technical specification LCO was not applicable due to the shutdown and defueled condition of the plant. The inspector agreed with the licensee's assessment and had no further question on log keeping. The inspector noted that the licensee was maintaining the fire system in an operable status even though not required by the technical specifications.

The inspector attended PORC meeting 86-02 on January 15, 1986 to review the several nonconformance reports brought before the committee for dispositioning. The inspector noted that each item was addressed in detail, followed by rigorous discussions of each discrepancy to assure causes were identified and corrective actions were appropriate. The PORC meeting minutes were reviewed and found to reflect the meeting discussions.

5.2 The licensee notified the inspector on January 16, 1986 that three valves on a line penetrating primary containment that require testing per Technical Specification 4.7.A.2 and Appendix J to 10 CFR 50 were never included in the primary containment leakage test program, and thus have not been subjected to a Type C test since plant startup in 1972. The valves are on a 1 1/2 inch diameter condensate line that passes through drywell penetration X-48 and provides a supply of condensate water inside the containment. Valve CST-11 is located on the outboard side of the penetration, and valves CST 11A and 11B are the inboard, parallel supply isolation valves to two hose connections. All three valves are specified by plant procedures to be closed for normal operations. The inspector noted that the valves were verified shut by plant operators per OP 2115 for operations during the last operating cycle.

The licensee determined this item is reportable under 10 CFR 50.73(a)(2)(ii) since a principal safety barrier (primary containment) was in an untested (unanalyzed) condition. An evaluation will be completed for the LER submittal. The condensate supply line is normally filled with water during routine plant operations. On January 12, 1986, the inspector discussed his concerns with a licensee representative regarding the need to assure no other containment penetrations have been omitted from the Type C program. The licensee stated that all other penetrations designated as a "spare" would be reviewed to assure all isolation valves are accounted for.

This item will be followed on a subsequent inspection pending submittal of the LER, and NRC review of the licensee's evaluation of the event (UNR 50-271/86-01-01).

5.3 During maintenance activities to refurbish the main steam isolation valves on January 16, 1986, plant technicians discovered that contact blocks to be used in the NAMCO limit switches were broken or chipped. All 35 contact blocks ordered per purchase order 20145 were inspected and 15 were found to have defects. Since no broken pieces of the damaged blocks were found in the packages, it was evident that the blocks were shipped in a damaged condition by the manufacturer. I&C personnel reviewed the purchase order records for the materials and noted that the items were received on site prior to implementation of the new receipt inspection program on August 1, 1985. NCR 86-07 was written to document the nonconforming conditions and to establish actions to disposition the defective parts.

The inspector examined the defected parts on January 24, 1986, and noted all 35 contact blocks were in the QC hold area of Stores. The defects consisted of cracks and chips in the insulating material of the contact block. Although the materials were of questionable quality, the size and location of the defects were such that no loss of function would have occurred if the blocks were used in the intended application.

This item is open pending completion of licensee actions under NCR 86-07, and pending NRC review of the receipt inspections performed by the licensee on the defected material (IFI 50-271/86-01-02).

5.4 The inspector reviewed radiation controls established by the licensee to verify that access control barriers, postings and surveys were appropriate for the work in progress. Controls established for radiation work permits (RWPs) 86-002, 645 and 651 were reviewed and found appropriate for the radiological hazards. The control of drywell work activities was reviewed routinely during either drywell entries or through the use of CCTV monitors established in the reactor building to verify workers did not linger or spend idle time in areas with high dose rates. Workers interviewed by the inspector were cognizant of the radiation fields in which they worked and the locations of low dose rate areas. No inadequacies were identified regarding licensee controls to maintain worker exposures as low as reasonably achievable.

During a tour in the drywell on January 23, 1986, the inspector noted at 3:15 P.M. that an airborne radioactivity area had been established at the 266 ft. elevation in the vicinity of work being done on valve RHR-46B per RWP 651. The inspector noted that the job site was controlled per the RWP requirements, but two of three access points to the area were not posted per 10 CFR 20.203(d) requirements for airborne radiation areas. An intended barrier rope for one of the access points had apparently fallen down. This matter was reported to health physics technicians covering drywell work who took actions by 3:20 P.M. to properly barricade and post all access points to the area. The inspector reviewed licensee radiation survey data in the work area and noted that the actual concentrations of radioactivity remained below 4.0X10-10 microCi/cc and thus, no airborne radiation area existed.

No violations were identified. The inspector had no further comments on this item at the present time. Posting of radiation areas will be reviewed further during subsequent inspections.

5.5 During post-maintenance testing of the reactor protection system (RPS) per OP 4318 on January 28, 1986, plant technicians noted anomalies while resetting the control rod drive system from a "scram" condition. All control rods were inserted and no fuel was in the reactor at the time. Following the satisfactory generation of a scram signal per step 7 of the procedure, the technicians noted that the scram condition on channel B would not reset with the mode switch in the "shutdown" position. The scram condition was reset by placing the mode switch in the "refuel" position. Investigation of the RPS circuits per drawings B191301 sheets 809, 816, 828 and 830 revealed that the scram reset problem was apparently caused by loose terminal screws on the line side of fuse F21B (CRP 9-17, terminal strip CC-97). An open circuit through F21B would prevent resetting a scram condition with the mode switch in shutdown, based on a loss of power to relays 5A-K28B and 5A-K29B.

The licensee tightened the fuse terminal screw 1-1/2 turns which apparently corrected the problem by energizing the K28B and K29B relays and allowing a reset of the scram condition. The system was scrammed manually and reset three times on January 28, 1986. No further action was taken on channel B since the problem was thought to be solved. A maintenance request was written and subsequently completed to check a sampling of terminal screws in the RPS cabinets for tightness. No further loose screws were found. A potential report form was written to evaluate the failure to reset condition for reportability under 10 CFR 50.73.

Subsequent to the above actions, the RPS channel B again failed to reset from a scram condition with the mode switch in the shutdown position. This item is open pending NRC review of licensee actions to (i) identify and repair RPS channel B; and, (ii) evaluate the incident for reportability to the NRC. The inspector stated to licensee management that the RPS operability problems should be corrected as a prerequisite to core reload activities. This item is unresolved (UNR 50-271/86-01-03).

5.6 The inspector interviewed licensee personnel and reviewed records regarding plans and procedures to remove the station service water system from service for up to three weeks to allow maintenance on service water valves not normally serviceable otherwise, and to allow the installation of welded attachments of various service water lines as part of the piping support modifications in progress per EDCR 84-402 for the seismic reanalysis program. The only significant heat loads in the present plant condition are the spent fuel pool cooling system and the station air compressors. Procedures were established to provide continued cooling to these loads via a mechanical bypass request that would supply river water to the reactor and turbine building closed cooling water heat exchangers using temporary piping and a feed from the fire water system.

Mechanical Bypass Request 86-008 and procedure OP 2181.1 were prepared to supply the alternate cooling to the spent fuel pool cooling system. A safety evaluation was completed in accordance with 10 CFR 50.59. The Plant Operations Review Committee reviewed the procedures and mechanical bypass. Yankee NSD also reviewed the bypass and safety evaluation and provided an independent review of the estimated heat loads. The inspector reviewed the licensee's procedures and evaluations and noted that consideration was given to the following items.

(i) The current spent fuel pool heat load and pool heat up rate was calculated and verified by testing prior to implementation of the bypass.

(ii) The bypass cooling flow to the RBCCW heat exchangers would be verified to be sufficient to cool the loads prior to release of the service water system for maintenance and modification work.

(iii) The electric fire water pump would be operated continuously. The normal power supply for the pump would come through the 345KV system, with the 115KV system and the Vernon tie line as backup sources. The diesel driven fire water pump could also be used as an alternate supply.

(iv) The electric fire water pump would be operated at less than 25% of its rated capacity and could be supplemented by the diesel driven pump should it become necessary to fight a fire. At no time would the flow requirements for fire system functional performance be degraded. An alternate cooling source using a feed and bleed of the spent fuel pool via the condensate storage tank would be available as a backup.

(v) The nature of the maintenance and modification activities would allow returning the normal service water system to service in less than 12 hours. A demonstrated heat up rate of less than 2°F per hour and an established pool temperature administrative limit of 125°F would assure that adequate cooling through the normal mode could be achieved prior to reaching the 150°F temperature limit.

(vi) Contingency instructions, action points and administrative limits were established in the operating procedure.

(vii) Requirements to sample and monitor the RBCCW effluent water were established.

Based on the above, the inspector concluded that the licensee's procedures and administrative controls were adequate to assure continued cooling of critical heat loads during the periods the service water system was shut down. No unreviewed safety questions were created by the proposed mode of operation. The licensee subsequently implemented the alternate cooling procedures and demonstrated the ability to maintain spent fuel pool temperatures at 120 degrees F or less, which was well below the technical specification limit of 150 degrees F. The resident inspector will follow system performance and licensee restoration actions on a subsequent routine inspection (IFI 50-271/86-01-04).

5.7 During baseline radiographic (RT) examinations of the N2D nozzle on December 29, 1985, the licensee identified two axial cracks in the old safe end to nozzle weld. The radiographs showed that the cracks were about 1 inch apart, started in the heat affected zone on the safe end side of the weld, and travelled at least half way into the weld material. The safe end material is considered susceptible to IGSCC. The weld was not included in the IGSCC examinations completed by the licensee per IE Bulletins 82-03 and 83-02. Open Item Report (OIR) 045 was issued to evaluate and repair the defect in preparation for installing the new safe end on N2D.

The licensee removed the defect by grinding from the ID side of the weld, and noted that the cracks travelled toward the stainless steel cladding on the nozzle and to within 0.125 inches of the surface (the material thickness was 0.75 inches, nominally). The inspector reviewed design drawing CB&I 9-6201 #22 Revision 7 and licensee calculations of the joint geometry to verify a minimum of 1/8 inch cladding thickness remained on the nozzle. The inspector completed an independent calculation of the cladding thickness, which confirmed the licensee's results. The inspector had no further questions on the N2D repair prior to installation of the new safe end.

During discussions with the Pipe Replacement Task Force Project Engineer on January 2, 1986, the licensee stated that a circumferential, volumetric defect was identified in the N1A nozzle to safe end weld. The defect (incomplete fusion) was identified during baseline RT of the joint preparation area, and occurred in the safe end to nozzle weld completed in 1968. Both N1 safe ends were replaced in 1970 since the materials were sensitized with the reactor vessel when it was furnace heat treated on January 20, 1969. The N2 safe ends were installed after vessel heat treatment from April 24 - May 21, 1969 (reference CB&I drawings 9-6021 R-3 Revision 3 and HT-2 Revision 2). The licensee does not have the RT records for the 1968 N1 welds. A re-examination of the RT records for the 1970 welds showed the defect identified in 1985, but outside the area of interest for the work done in 1970. The defect was removed to prepare the joint for the new safe end installation.

The inspector had no further comment on this item at the present time. The task force issued a potential reportable form (PRF) describing the N1A and N2D defects, which was forwarded to the Engineering Support Department for review and dispositioning. The item will be followed on a subsequent inspection pending completion of the licensee's evaluation of the defects for reportability (IFI 50-271/86-01-05).

5.8 During a weld prep rework for the N2F nozzle on January 14, 1986 per work package 1420, carbon filler material ER70S-3 was used rather than the required ER 308L weld wire. The carbon steel weld wire was issued in error by the containment access rod room attendant, and the mistake was not detected by the welder or the QC personnel who checked the weld. The inspector interviewed the M+K and Task Force QA Managers and reviewed OIR-055, NCR-058 and NCR-058-S, which described the circumstances of the event and the licensee's evaluation and corrective actions for the faulty rework and cause of the error.

A welder submitted material withdrawal slip to the attendant on January 14, 1986 after obtaining the welding supervisor's approval to do the work in accordance with welding procedure M88A. A weld material withdrawal slip issued by the welding supervisor per WE-1 specified the use of bare ER308L material. Upon presentation of the slip, the attendant mistakenly issued coated carbon weld wire after recording heat number W05-001 on the form. The welder noted that coated wire was an error and requested bare wire. The attendant filled in a new material withdrawal slip (#05036) by transcribing information (including the erroneous carbon heat number) from the original, discarded the original slip, and signed the new form for the foreman. The welder accepted the bare wire and did not check the material type and heat number tag on the wire further since he reportedly did not expect a second mistake to be made.

The QC inspector at the job site verified that the welder was qualified to do the rework per an approved procedure, and entered the welder ID and weld wire heat number on the weld data card. Since the heat number was taken from the withdrawal slip, the QC inspector failed to note that the material did not conform to the heat number (W02-001) for type ER-308L wire. The welder completed the rework and returned the unused material to the rod issue room, which was then staffed by a new attendant after shift change. The new attendant noted that the returned wire stubs did not match the withdrawal slip. Further work on N2F was suspended pending review and resolution of the discrepancy.

The inspector reviewed the licensee's followup for this item which included the following actions.

(i) the welder and the attendant were terminated.

(ii) all carbon steel weld wire was removed from the containment access rod room since it is currently not needed for drywell work.

(iii) weld wire withdrawal slips for manual welding in the drywell were reviewed to assure none were completed in the same hand writing, and all were properly completed by the foreman and the attendant.

(iv) an inventory of all E70S-3 electrodes was completed to account for all electrodes.

(v) all weld data cards for drywell work were reviewed to verify that none contained a E70S-3 ID number (W05/001).

(vi) procedures were revised to require QC verification of the material at the point of issue and at the weld joint. Inspectors were instructed to record information from the material rather than the slips.

(vii) blank material withdrawal slips were removed from the rod rooms and issued only to welding foremen and supervisors.

(viii) welders and attendants were retrained on the proper use of material withdrawal slips.

Based on the above, the inspector concluded that licensee actions were sufficient to assure no other improper welding with carbon steel wire had occurred in the drywell, and actions were sufficient to preclude a recurrence of the event.

No violations were identified. The inspector had no further comments on this item. This issue was reviewed further in NRC Region I Inspection 86-02.

5.9 During an inspection tour inside the drywell on January 23, 1986, the inspector reviewed the general quality of welding on structures and components important to safe operation of the reactor. Welding on the following was satisfactory: structural steel at the 269 ft. elevations, 45 degrees to 135 degrees; piping deadweight supports and hangers MS-23, MS-34, FW-20, MS-58, MS-59, MS-35, MS-21, MS-14, MS-5, FW-9, HPCI-3, and MS-18; upper and lower core spray headers from 45° to 135°; restraining steel around the main steam and feedwater lines at the 269 ft. and 252 ft. elevations; and, tube and structural steel supporting the pipe break energy absorption panels protecting the drywell liner.

One item that warranted further review was the acceptability of support steel welding at two locations on the energy absorption panels. The questionable areas were reviewed with a licensee representative on January 23, 1986, and compared against drawings 5920-FS-1557. The locations of interest had toe welds where full welds were required. This item will be reviewed further on a subsequent inspection pending completion of licensee review and dispositioning of the discrepancies (UNR 50-271/86-01-06).

5.10 During a meeting with the Maintenance Superintendent on January 15, 1986, the licensee stated his intention, pending concurrence from NRC Region I, to discontinue the fire watch established for areas on the 232 ft., 252 ft. and 280 ft. elevations that were found deficient in meeting Appendix R requirements. The basis for ceasing the roving patrols was that any hot work in the areas would be covered by fire watches established per AP 0042, and due to the fact that no equipment in the designated areas was required to be operable in the current plant condition. Following consultation with NRC Region I personnel, the inspector concurred with the licensee's actions and noted that the Appendix R fire watches would be re-established prior to declaring the plant systems in the affected fire areas operable to support reactor refueling.

Since the commitment to maintain the fire watches was established in FVY 84-149 dated December 28, 1984, the inspector stated that the licensee should document his intention in a letter to NRC Region I. The licensee acknowledged the inspector's comments. This item is open pending receipt of the licensee's letter, and pending review of licensee actions to either fix Appendix R deficiencies in the affected plant areas, or re-establish fire watches (IFI 50-271/86-01-07).

5.11 The licensee made a 4-hour notification to the NRC on January 28, 1986, per 10 CFR 50.72(b)(2)(iii) regarding the potential degradation of the primary containment hydrogen/oxygen analyzer. A recent review by YNSD engineering identified an error in the dose rate calculation used to establish environmental qualification for the monitors. The error apparently underestimated the integrated dose that viton seals used in both redundant trains would be exposed to in the post accident environment. The monitors are relied upon to mitigate a design basis accident. It appears that, if inoperable, the monitors would have been incapable of performing the intended safety function ever since both trains were first declared operable in the Fall of 1984. The licensee's evaluation of the impact of the error on system operability was in progress at the conclusion of the inspection.

This item is unresolved pending completion of the licensee's evaluation of system operability and subsequent review by the NRC (UNR 50-271/86-01-08).

5.12 The licensee notified the inspector on 1/24/86 that I&C technicians had identified that the sensing lines were crossed on pressure switches PS-2-128A and 128B. The switches are used in the shutdown cooling permissive circuitry in the RHR system. The licensee reported that since both sensing lines tap off the suction side of the "B" recirculation pump, and since the switch output contacts are arranged in a series configuration for the shutdown cooling permissive, the crossed sensing lines had no impact on system function. The licensee initiated actions to correct the error.

This item is open pending completion of licensee actions to correct the sensing line routing, and pending further NRC review to verify the piping and circuit arrangement (UNR 86-01-09).

6.0 Outage Maintenance and Testing

6.1 The inspector reviewed surveillance test OP 4217, "Alternate Shutdown Battery AS-2 Service Test", conducted January 14, 1986, to verify compliance with procedural requirements. The AS-2 battery is one of two batteries that comprise the Alternate Shutdown Battery System. OP 4217 involves discharging the batteries the same amp hours as would be experienced during the worst case loading conditions of DC bus DC-2AS.

The inspector reviewed the following items: test data demonstrates conformance with Technical Specification requirements; test results were reviewed and approved by supervisory personnel; test equipment in service properly calibrated, and, restoration of the system to normal operating condition.

No inadequacies were identified.

6.2 The inspector observed portions of selected safety related maintenance activities to verify that the activities were being conducted in accordance with approved procedures. The following items were considered during the review of maintenance activities: the procedures used were adequate to control the activity; activities were being accomplished by qualified personnel; replacement parts and materials were properly certified; and QC hold points were established and observed.

+ MR 85-2255: repair MOV 23-19 (HPCI) seat leakage
+ MR 85-1796: repair MOV 13-21 (RCIC) seat leakage
+ MR 85-1942: repair MOV 2-77 (MS) seat leakage
+ MR 85-1711: repair MOV 14-12B (CS) seat leakage

The items below warranted followup by the inspector.

Main Steam MOV 2-77 and Core Spray MOV 14-12B failed local leak rate testing due to high seat leakage. Investigation of MOV 2-77 discovered cracks in the Stellite-6 face of the gate valve 'wedge', and arrangements were made to reface the wedge with stellite material. The wedge of MOV 14-12B was found to be undersized and riding on the guides. A vendor was contracted to perform the above mentioned work. The inspector interviewed the contractor and noted the individual was knowledgeable of the procedure, materials, and repair process. The inspector also reviewed the contractor's welding procedures, equipment, and qualifications, and noted the following discrepancies: the calibration on the thermocouple used in the stress relieving oven had recently expired; the documentation certifying the welder was qualified to perform the work was filled in, but not signed; and, one procedure which the welder intended to use for the work had not been reviewed and approved by the licensee.

These items were brought to the attention of the licensee and contractor for correction. The contractor had not started any welding on either valve. The licensee postponed the job until the proper procedures were approved, welder certification records were accurate and complete, and, the stress relieving oven received a proper calibration.

The inspector reviewed the licensee's corrective actions and found them appropriate. The inspector expressed his concerns with licensee management regarding the apparent failure by VY staff presonnel to identify the aforementioned QA deficiencies associated with the job. This item is unresolved pending further NRC review of contractor work activities for the outage (UNR 86-01-10).

7.0 Review of Outage Modifications

The inspector reviewed licensee activities in progress during this outage to upgrade seismic supports on plant systems in accordance with EDCR 84-402. The seismic reanalysis program is a licensee initiative that has been in progress over the last several years to upgrade the plant seismic design to present day analysis criteria. However, the licensee's analysis of baseplate flexibility to resolve concerns raised in IE Bulletin 79-02 is also incorporated in the seismic reanalysis effort. The licensee committed to finish actions in regard to IEB 79-02 prior to startup from the current refueling outage, in accordance with the requirements of Region I Confirmatory Action Letter 85-06. A review of the licensee's implementation of EDCR 84-402 in accordance with the programmatic requirements of AP 6000 was begun on this inspection and will be continued during subsequent routine inspections.

The major effort completed in this area for this inspection consisted of reviewing the as-built conditions for a sample of modified supports to assure the completed work met the design, and to assure the adequacy of the licensee's modification controls. The supports reviewed during this inspection are identified below, and additional supports will be reviewed during future routine inspections. The inspector determined through direct observation and independent evaluation of work that the licensee's work control system was functioning properly and that the installation of safety-related pipe supports and restraints was in compliance with NRC requirements, licensee commitments and applicable codes. The inspector verified the following: weld location; the surface of welds did not contain discontinuities, abrupt ridges, valleys, undercuts, cracks, or other detrimental indications; supports were located and installed as specified on the drawing; support clearances were as specified (includes support to pipe and base plate to ceiling/wall clearances); bolts, nuts, washers properly sized; proper bolt location; minimum bolt embedment and thread engagement requirements satisfied; and overall dimensions.

The inspection included the following hangers, for which construction had been completed and as-built conditions had been verified by the contractors QC group:

ACSP-HD33K	RHR-H160	RHR-H166
CST-H19	RSW-H264	RSW-H254
CST-H20	RSW-H208	RHR-HD201B
HPCI-HD74B		

The following problems were noted and discussed with the licensee representatives for resolution.

CST-H19 Discrepancies between as-built hangers in the CST pipe trench and the isometric/hanger details.

RHR-H160 Excessive space (greater than 1/16") between baseplate and ceiling. Condition evident over 40% of the edge of the plate.

RSW-H264 Discrepancies noted on the welding of a shim, which met the requirements of the general notes but not the as-built drawing.

HPCI-HD74B Excessive space (approximately 1/8") between base plate and wall.

This item is open pending review of the licensee's corrective actions on the above hangers (UNR 86-01-11).

Additionally, during routine tours the following hangers received a general review:

CST-H17	CST-HD47A	CST-HD17A
CST-H46	CST-HD47B	CST-HD19A
CST-H47	CST-HD47C	

These hangers were reviewed to the same criteria stated above but without the aid of a detailed hanger drawing. No discrepancies were noted.

A detailed review of records maintained on several of the hangers documented above included:

+ Weld identification/location documented by QC
+ Welding material used corresponds to the material specified
+ Welder qualified to the welding procedures used
+ Torque wrenches properly controlled and calibrated
+ Personnel engaged in QC inspection properly trained to perform their specific tasks

No discrepancies were noted.

8.0 Followup of Previous Inspection Findings

8.1 Followup of Item 85-25-06. The licensee notified the inspector on January 10, 1986 of his intention to submit a Part 21 (reference FVY 86-3) report regarding the degradation observed in the General Electric containment electrical penetrations previously reported in LER 85-10 dated October 30, 1985. As reported in LER 85-10, licensee inspections of penetrations containing control and indication circuits determined that: conductors were installed such that some were in contact with the sharp edge of the end of the penetration sleeve; and, insulation damage could occur, as evidenced by a short identified in penetration X105C. There are no edge protectors installed on the edge of the penetration assemblies. Over time, the weight of the large number of conductors in the center of the penetration causes the sharp edge of the penetration sleeve to cut into the insulation of wires located on the bottom of the penetration. Four of six penetration types (control and indication, 480 volt power, neutron monitoring, and control rod position indication) were found susceptible to the failure mechanism. Inspection of the reactor building side of the 5KV power and thermocouple penetrations found no conductors in contact with the metal edge.

Corrective actions in progress include abandonment of the lower penetration wires and relocation of control circuits to conductors in the center of the assemblies. Although no specific failures have been identified that could have adversely affected safe plant operation, the licensee determined that a potential safety impact existed in that safety related circuits use wires near the bottom of the penetration assemblies affected by the failure mechanism. Additionally, although no safety circuits have failed to date, the possibility exists that some damage has occurred that would only be observable under more severe accident conditions.

Five of seven installation and test procedures needed to implement the corrective actions have been completed, and work activities are expected to start in Feburary, 1986. The resident inspector will follow the licensee's corrective actions.

8.2 Followup of Violation 84-08-06. Vermont Yankee letter FVY 85-61 dated July 1, 1985 presented for staff review the Summary Report for the Detailed Control Room Design Review (DCRDR) completed in accordance with the NRC Confirmatory Order dated June 12, 1984 and per NUREG 0737, Supplement 1. The DCRDR identified 54 human engineering discrepancies (HED), which were summarized by categories in the report. Section VI of the summary report, redefined Class A HEDs as those which generally relate to the EOPs or the Technical Specifications, and Class B HEDs as those findings that have the potential to cause human error or equipment misoperation. The inspector noted that none of the 54 HEDs apparently addressed the concerns raised by an event which occurred on April 20, 1984 (see LER 84-05) wherein the HPCI initiation logic on High drywell pressure was blocked by a failure to reset the high vessel water level isolation signal following a scram on April 16, 1984. The existing design at Vermont Yankee provides no HPCI logic status information to alert the operator that system initiation is blocked.

During a meeting with the DCRDR Project Manager on February 3, 1986, the inspector noted that the concerns identified in LER 84-05 were addressed as Finding 0661 in the listing of 863 potential HEDs identified by the DCRDR. Finding 0661 was grouped along with other similar findings and summarized as an annunciator deficiency as Class A HED 0602. The final resolution of HED 0602, along with the HPCI item, remains to be dispositioned by the DCRDR Management Team.

This item remains open pending completion of licensee actions to resolve the concerns with the HPCI logic status information and further review by the NRC.

9.0 Procedures for Degraded Grid Conditions

The inspector met with the Operations Superintendent and the Operations Supervisor regarding the status of actions to implement alarm response procedures per OP 3140 for a "low grid voltage without an accident signal" condition. The proposed procedures were reviewed and concurred with by the NRC staff during inspection 85-29. The licensee stated that revised procedures to address this item would be approved and issued prior to startup from the current modification outage. This item will be reviewed further on a subsequent inspection to verify the operating procedures are revised per the above commitment (IFI 86-01-12).

10.0 Annual Retraining

The inspector attended an annual retraining lecture on January 2, 1986, as part of the licensee's program to meet site access requirements. The inspector noted that the presentation materials and lectures on health physics, security and safety were of high quality and exceeded minimum acceptable requirements. Minor discrepancies in the presentation were discussed with training representatives, who noted the comments for consideration in future updates of the presentation materials. No inadequacies were identified.

11.0 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.