U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No.	91-12
Docket No.	50-271
Licensee No.	DPR-28
Licensee:	Vermont Yankee Nuclear Power Corporation RD 5, Box 169 Ferry Road Brattleboro, VT 05301
Facility:	Vermont Yankee Nuclear Power Station Vernon, Vermont
Inspection Period:	April 21 - June 1, 1991
Inspectors:	Harold Eichenholz, Senior Resident Inspector Thomas G. Hiltz, Resident Inspector Peter D. Drysdale, Senior Reactor Engineer Alan E. Finkel, Senior Reactor Engineer Alfred Lohmeier, Reactor Engineer
Approved by:	John F. Rogge, Chief, Reactor Projects Section 3A — 6/26/91 Date

Inspection Summary: This inspection report documents routine resident safety inspections conducted between April 21 and June 1, 1991. Station activities inspected during this period included: plant operations; radiological controls; maintenance and surveillance; emergency preparedness; security; engineering and technical support; and safety assessment and quality verification.

Results: Inspection results and conclusions are summarized in the attached Executive Summary.

EXECUTIVE SUMMARY
Vermont Yankee Nuclear Power Station
Report No. 50-271/91-12

Plant Operations

A loss of off-site power event occurred which resulted in the dispatch to the plant of an Augmented Inspection Team. Subsequent start-up and power ascension activitie were conducted by control room personnel in a professional manner. Reviews were conducted of events involving Primary Containment Isolation System (PCIS) Group VI actuation (LER 91-11), the tripping open of the reactor protective system "A" motor-generator output breaker and subsequent PCIS Group III isolation, and the automatic closure of the reactor water clean-up system isolation valve (LER 91-10). No safety concerns were identified and the Vermont Yankee corrective actions were appropriate. A previously identified NRC concern (VIO 50-271/90-09-01) involving valve lineup deviations being conducted without 50.59 evaluations and Plant Operations Review Committee reviews was closed. The adequacy of the supports and restraints for the carbon dioxide system's high pressure gas cylinders located in the cable vault and switchgear rooms remained unresolved (UNR 50-271/91-12-02).

Radiological Controls

A review of an event involving an unsecured radioactive source was conducted. The Vermont Yankee event evaluation was not sufficient and warranted further licensee and NRC review. The approval of the Radiation Protection Program Improvement Plan, and the inclusion of the Radiation Protection (RP) Department in the personnel Rotation Development Program were positive accomplishments.

Maintenance and Surveillance

Maintenance activities involving diesel fi pump Limiting Condition for Operation (LCO) maintenance and standby liquid control pum corrective maintenance were properly planned and well controlled. The Vermont Yankee actions to remove this equipment from service to accomplish maintenance activities demonstrated a prudent approach to plant safety. The adequacy of the corrective actions associated with recurring deficiencies in the visual examination process remained unresolved (UNR 50-271/91-12-03). The surveillance testing program was effectively being implemented and identified equipment deficiencies were being properly corrected.

Emergency Preparedness

An Unusual Event was declared on April 23 as a result of a loss of off-site power. The Vermont Yankee Emergency Plan was effectively implemented during this Unusual Event.

Security

An incident involving a non-specific threat was responded to in a conservative and professional manner. The relocation to Gatehouse No. 3 as a main personnel entranceway into the Protected Area and Vermont Yankee response to an anti-nuclear demonstration demonstrated well planned and controlled activities by Vermont Yankee and contractor security personnel. Communications and coordination between Vermont Yankee and local law enforcement authorities continues to be a notable strength.

Engineering and Technical Support

The review of the environmental qualification concerns due to unanalyzed heating steam lines (LER 91-08) determined that corrective actions were appropriately identified. A review was conducted by an NRC Region I specialist inspector of an NRC program to characterize a flaw in a previously removed from service feedwater check valve. An enhancement to the Emergency Response Facility Information System was identified as a notable performance strength.

Safety Assessment and Quality Verification

Three Licensee Event Reports (LERs) adequately characterized the subject operational events.

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DETAILS

1.0 SUMMARY OF FACILITY ACTIVITIES

Vermont Yankee Nuclear Power Station (VY) began the inspection period operating at 100 percent of rated power. On April 23, the reactor automatically shutdown from 100 percent of rated power due to a loss of off-site power caused by maintenance activities being performed on switchyard batteries which provide control power to the switchyard breakers. An Unusual Event (UE) was declared and a number of 10 CFR 50.72 required notifications were made to the NRC as a result of this event. NRC Region I (NRC:RI) dispatched a five member Augmented Inspection Team (AIT) to the plant to review this event. The results of the AIT are documented in NRC Inspection Report 50-271/91-13. Subsequently, a plant start-up occurred on April 30 and the plant returned to full power operations on May 1.

Throughout the remainder of the period, short term scheduled power reductions to 97 percent of rated power were conducted weekly to perform routine surveillances on control rod drives, main turbine valves and by-pass valves. On May 19, a decrease to 80 percent of rated power occurred to facilitate a rod pattern exchange. Off-gas activity levels of between 18,000 to 29,000 uci/sec occurred during the period and analysis indicated that this activity level was primarily due to recoil effects from fuel failures experienced during the previous cycle. As of June 1, the close of the inspection period, off-gas activity was at 22,000 uci/sec and the plant was operating at full power.

During this assessment period, site characterization work was conducted within the Owner Controlled Area of VY. This work consisted of digging test pits and drilling test wells. These efforts are part of the low-level radioactive waste facility studies being conducted to identify potential repository sites in the State of Vermont.

2.0 PLANT OPERATIONS (71707, 93702, 92701, 92702, 62703, 61726)

2.1 Inspection Activities

The inspector verified that the facility was operated safely and in conformance with regulatory requirements. Management control was evaluated by direct observation of activities, tours of the facility, interviews and discussions with personnel, and independent verification. The inspector performed backshift inspections including deep backshift, weekend, and holiday inspections on April 23, 24, 25, 27, 30, and May 19 and 21.

2.2 Inspection Findings and Significant Plant Events

2.2.1 Loss of Off-Site Power/Unusual Event

On April 23, at 2:48 p.m., VY lost the capability to supply or receive 345 kV and 115 kV off-site power. The cause of the loss of off-site power was related to maintenance activities associated with connecting battery DC-4A to its DC control power bus. Several breaker failure relays failed and initiated a complex switchyard breaker opening sequence, effectively isolating

the plant from normal off-site power sources. The turbine generator was automatically removed from service and an automatic reactor shutdown resulted from main turbine control valve fast closure. All control rods fully inserted on the scram as designed.

At approximately 3:07 p.m., the operating crew Shift Supervisor declared an UE due to zero voltage indication on the 345 kV and 115 kV electrical distribution buses. Notifications to the States of Vermont, New Hampshire and the Commonwealth of Massachusetts were completed within 15 minutes from the declaration of the UE. Off-site power through the 115 kV switchyard was restored by approximately 7:25 p.m. on April 23. Off-site power from the 345 kV switchyard was restored at approximately 4:17 a.m. on April 24, through a main transformer backfeed electrical alignment. The delay in restoring power via this path was due to the greater than anticipated effort needed to remove the disconnect links from the generator output conductors. The UE was terminated at approximately 7:50 p.m. on April 24.

A five member AIT arrived at VY on April 25 to: (1) determine the circumstances and events which led to the loss of off-site power, (2) evaluate VY's actions in response to the event, (3) evaluate the response of the plant systems to the event, and (4) determine if any potential generic issues were associated with the event. The Vermont State Nuclear Engineer, at the invitation of the NRC, observed AIT inspection activities.

The Vermont Yankee Emergency Plan was effectively implemented during this UE. The loss of off-site power did not threaten the public health and safety. Detailed information concerning the VY loss of power on April 23 is contained in the AIT Inspection Report 50-271/91-13.

2.2.2 Reactor Start-up Following the April 23 Loss of Off-Site Power

At approximately 1:57 a.m. on April 30, control room operators commenced a reactor start-up. This start-up followed an approximately 7 day unplanned outage which resulted from an automatic reactor shutdown and loss of off-site power on April 23. The reactor start-up commenced after an extensive review and effective resolution of the safety concerns which resulted from the loss of off-site power event. These safety concerns centered on assuring switchyard battery system reliability and adequate service water cooling to plant equipment.

Control room operators declared the reactor critical at approximately 3:00 a.m. on April 30. A Primary Containment Isolation System (PCIS) Group VI isolation, actuated during plant heat-up operations, was expeditiously reset. The PCIS Group VI isolation did not result in a plant transient and the reactor remained at power. This event is discussed in Section 2.2.3 below.

The inspector observed Control Room activities from the commencement of the reactor start-up to criticality. The inspector also observed Control Room activities during main generator synchronization to the power distribution grid and periodically observed Control Room activities during reactor power ascension. Control room activities were professionally conducted. Licensed operators routinely referred to appropriate procedures and demonstrated a good questioning attitude.

2.2.3 **PCIS Group VI Isolation During Reactor Start-Up (LER 91-11)**

At approximately 5:55 a.m. on April 30, with a plant heat-up in progress, PCIS Group VI isolation actuated. The PCIS actuation isolated two steam supply valves in the high pressure coolant injection (HPCI) system and two steam supply valves in the reactor core isolation cooling (RCIC) system. At the time of the PCIS Group VI isolation, control room operators were returning the PCIS Group VI isolation low steam supply pressure trip functions to service. These PCIS Group VI isolation trip functions are normally bypassed when reactor pressure drops below 150 psig to prevent unintended actuation of the PCIS. The procedure for returning these trip functions to service requires that the trip functions be returned to service prior to reactor pressure exceeding 150 psig; however, the procedure does not specify a minimum reactor pressure above which the trip function can be restored without incurring a PCIS Group VI isolation. Following the isolation, operators returned the bypass switches to the bypass position and opened the HPCI and RCIC steam supply valves. Reactor power remained at approximately 2 percent of full rated power and was not affected by this event.

According to the plant Technical Specifications (TS), the PCIS Group VI isolation setpoint is required to be greater than or equal 50 psig for the RCIC system and greater than or equal to 70 psig for the HPCI system. The reactor pressure at the time of the event was approximately 116 psig. When reactor pressure was approximately 132 psig, the PCIS Group VI isolation trip functions were successfully returned to service.

VY reported this event to the NRC in accordance with 10 CFR 50.72. On May 28, VY submitted Licensee Event Report (LER) 91-11 describing the event details and their evaluation.

VY determined the root cause of this event to be an incomplete procedure in that the procedure did not provide a minimum reactor pressure that must be achieved prior to returning the PCIS Group VI isolation trip functions to service. VY's corrective actions for this event included a procedural revision and subsequent procedural training for licensed operators.

The inspector determined that, pending evaluation of the as-found PCIS Group VI isolation trip setpoint, the VY root cause determination, event analysis, and corrective actions associated with this event were adequate. The inspector concluded that this event was of minor safety significance.

2.2.4 **Reactor Protection System (RPS) "A" Motor-Generator Output Breaker Opening**

On May 28 at approximately 8:39 a.m., the RPS "A" motor-generator output breaker opened. The RPS "A" motor-generator is the normal power supply to the "A" RPS electrical bus. When the RPS motor-generator output breaker opened, the "A" RPS bus de-energized and a half scram signal was received. In addition, power to two reactor building process radiation monitor high radiation relays is supplied from the "A" RPS electrical buses. When these relays de-energized a PCIS Group III isolation occurred resulting in a reactor building ventilation (RBV) shutdown and a standby gas treatment system automatic start.

The power supply to the "A" RPS electrical bus was transferred to the alternate power source and at approximately 8:50 a.m., the Senior Control Room Operator notified the NRC Operations Center of the PCIS Group III actuation in accordance with 10 CFR 50.72(b)(2)(ii). The cause of the output breaker failure was not immediately determined and the RPS "A" motor-generator was declared inoperable to support maintenance troubleshooting.

The RPS "A" motor-generator was operated for approximately 24 hours with the output breaker shut and the RPS "A" motor-generator bus supply breaker open. The output breaker remained closed during this troubleshooting period. After extensive maintenance troubleshooting, the root cause for the output breaker opening could not be determined. The RPS "A" motor-generator was placed in-service on May 29.

Licensed control room operator response to the event appeared to be good. The equipment failure was accurately diagnosed, the power supply to the "A" RPS electrical bus was expeditiously transferred, the PCIS Group III isolation reset, and RBV restored to the normal line-up. The reactor remained at 100 percent of rated power and power operations were not affected by loss of power to the "A" RPS electrical bus. The maintenance activities appeared thorough and, although a root cause was not determined, they provided reasonable assurances that the RPS "A" motor-generator would function as designed.

2.2.5 **Reactor Water Clean-up (RWCU) System Valve 12-15 Automatic Closure (LER 91-10) (NCV 91-12-01)**

On April 12, with the reactor operating at 100 percent rated power, RWCU-15 valve closed, isolating the RWCU system suction from the reactor recirculation loop. RWCU-15 valve is the inboard PCIS Group V valve in the suction line for the RWCU system. The cause of the valve closure was a failure of the 16A-K26 relay coil which actuates closure of the valve. Relay 16A-K26 is a General Electric (GE) CR 120 Relay and a PCIS component. Electrical interlocks on the RWCU suction valve resulted in an automatic shutdown of the in-service RWCU pump. The relay coil was replaced, an appropriate system retest completed, and the RWCU system returned to service approximately 5.5 hours after the relay failure. Equipment affected by this event operated as expected.

VY determined the root cause of this failure was lack of an established service life for normally energized GE CR 120 relays. Based upon this failure and recent equipment history, VY established a service life of fifteen years for these relays. The failure of the coil is age related and, according to VY, the majority of the GE CR 120 relays used in the PCIS have been in service for many years and are approaching the end of their useful life. At Plant Operations Review Committee (PORC) meeting 91-36, the PORC expressed concern about the failure rate for this type of relay. Since October 1990, four GE CR 120 relays have failed. The I&C Department, in conjunction with Yankee Nuclear Services Department (YNSD), examined all remaining in-plant relays of this type. These examinations did not identify additional problems.

The inspector concluded that the safety significance of this singular event was minimal. The PCIS system functioned as designed and the relatively short time that the RWCU system was out-of-service did not adversely affect primary plant chemistry. However, the inspector shared the concern expressed by PORC about the failure rate for this type of relay. Corrective actions to replace all original GE CR 120 relays coils used in the PCIS appear appropriate.

This event was reported to the NRC in LER 91-10, "Failed Relay Coil Results in Primary Containment System Actuation." The event was determined to be reportable after conducting an engineering evaluation of the operational event. The determination that this was a reportable event in accordance with 10 CFR 50.73 was made by plant management on April 30. However, Engineered Safety Feature (ESF) actuations (PCIS is an ESF system) are also required to be reported to the NRC within four hours in accordance with 10 CFR 50.72. VY did not report this ESF actuation to the NRC in accordance with 10 CFR 50.72. This NRC identified reporting violation was of very minor safety significance and corrective actions taken by VY were appropriate. Since this violation meets the requirements of Section V.A. of the NRC Enforcement Policy, this violation was not cited (NCV 50-271/91-12-01).

2.2.6 **(Closed) Violation 90-09-01: Valve Lineup Deviations Conducted Without 10 CFR 50.59 Evaluations and PORC Reviews**

On October 29, 1990, VY responded to the Notice of Violation in NRC Inspection Report 50-271/90-09. VY proposed corrective actions that were intended to clarify their response to off-normal conditions and rectify identified implementation weaknesses. However, following an NRC meeting with VY on November 19, 1990, the NRC responded on December 11, 1990 that the VY proposal and their stipulated corrective actions were unacceptable and a revised response was appropriate. VY subsequently responded on January 10, 1991, stating that procedure revisions would be made to AP 0155 by March 1, 1991 that would be responsive to the NRC concerns. Inspection report 50-271/91-07 documented an NRC granted time extension until April 1, 1991 as the date for completion of the procedure revision.

The specific corrective actions committed to by VY consisted of: (1) provide the necessary guidance for determining when a 10 CFR 50.59 evaluation is required as part of conducting a valve or breaker lineup deviation; and (2) to require that, if it is determined that a 10 CFR 50.59 evaluation is applicable, the evaluation will be performed prior to implementation of the change. All licensed operators are to be trained in the intent and conduct of 10 CFR 50.59 evaluations by the end of the first cycle of the 1991 Licensed Operator Requalification (LOR) period.

On April 1, administrative procedure AP 0155, Rev. 14, "Current System Valve and Breaker Lineup and Identification," was reviewed and approved for implementation by the Plant Operations Review Committee (PORC). The inspector reviewed this revision and determined that it contained appropriate screening criteria to aid the Shift Engineer and Shift Supervisor in determining the need for a 10 CFR 50.59 evaluation. This criteria was developed by VY, in part, using industry document NSAC-125, "Guidelines for 10 CFR 50.59 Safety Evaluations."

To prevent the implementation of a lineup deviation without a completed and approved safety evaluation, if one were required by 10 CFR 50.59, the revised procedure AP 0155 specifies alternative administrative controls to allow appropriate shift personnel response to an off-normal equipment condition. Specifically, in this case, the implementation of a lineup deviation could be implemented by white tagging the component in the desired position, declaring the component inoperable and evaluating this condition under plant equipment control protocols (e.g., conform to TSs Limiting Conditions for Operations, if appropriate), and have a subsequent review conducted by the Operations Supervisor. This process does not preclude having a completed and approved safety evaluation available prior to implementation of a lineup deviation instruction. The stipulated controls are consistent with views expressed in the NRC letter to VY on December 11, 1990.

The end of the first cycle LOR occurred on May 31. The inspector reviewed Training Department lesson plans and documentation. Training for all licensed operators was conducted in accordance with VY commitments. Additionally, the Operations Department developed an adjunct instructor guide: "Valve and Breaker Lineup Deviations" and provided training relative to the issue and the newly revised procedure to all Senior Reactor Operator licensed control room personnel and to Shift Engineers. Interviews conducted with control room operators demonstrated that they were knowledgeable of the issue and the revised procedure requirements. A number of completed VYAPF 0155.03, "50.59 Safety Evaluation Screening Criteria" forms were reviewed. No concerns pertaining to applying the screening criteria were identified. This violation is considered closed.

2.2.7 **Carbon Dioxide System Bottle Supports and Restraints (UNR 91-12-02)**

The cable vault and switchgear rooms at the plant are protected by fully automatic total flooding carbon dioxide suppression systems. As part of these systems, high pressure (approximately 700 psig) gas bottles are installed in the cable vault and switchgear rooms. A mixture of various supports and restraints for the in-place storage of these bottles are in use in each area.

A tour of the plant was conducted by the inspector and a member of the Advisory Committee on Reactor Safeguards on May 23. During this tour, an issue was raised pertaining to the mixture of supports and restraints used to secure the bottles in-place in the cable vault. The specific concern related to the ability of some of the supports and restraints to properly secure the bottles in the cable vault during a seismic event.

VY was requested by the inspector to ascertain to what extent the bottle support and restraint system in these areas were analyzed and/or designed to withstand in a seismic event to preclude the development of a compressed gas cylinder missile hazard in areas containing safety related equipment. Subsequently, the Fire Protection Coordinator indicated to the inspector that VY did not believe that the cable vault fire suppression system was designed to withstand a seismic event. However, the safety evaluation for Plant Design Change Request No. 79-06, Switchgear/Cable Vault Suppression System, included a statement that the presence of high pressure gas bottles in the switchgear room does not sacrifice the integrity of equipment in this

area in light of the supports and restraints provided to store these bottles. VY was unable to provide the inspector with an analytical basis to support this statement. A walkdown on the subject system by VY engineering personnel to ascertain the nature and extent of the seismic withstand capability concerns was conducted. They concluded that it would be prudent to conduct further engineering analyses to determine the susceptibility of these systems to the postulated concern. Subsequently, the site engineering organization made a verbal service request to YNSD for engineering services to resolve this matter.

VY's initial response to this issue by the site engineering organization was appropriate. Engineering expertise was made available in a timely manner to enable VY to properly assess the NRC concerns. A deliberate and timely approach to the NRC concern was apparent. The adequacy of the support and restraint of the cardox system high pressure gas cylinders remains an unresolved item pending completion of the licensee's engineering review (UNR 50-271/91-12-02).

3.0 RADIOLOGICAL CONTROLS (71707)

3.1 Inspection Activities

Compliance with the radiological protection program was verified on a periodic basis.

3.2 Inspection Findings and Review of Events

The inspector frequently toured the Radiological Controlled Area and inspected several radiation work permit designated areas. During these tours the inspector assessed the effectiveness of the radiological housekeeping program, reviewed radiological posting requirements, and observed radiological work practices. The inspector found workers adhering to established radiological work practices.

3.2.1 Unsecured Radioactive Source

On March 29, a radioactive source was left in the Radiation Protection (RP) Calibration Lab with the cover off the shielded source container. The dose rate approximately 12" above the container was approximately 500 mr/hr. The area was not posted as a "High Radiation Area."

The inspector reviewed the VY evaluation of this event, discussed the event with the individual who left the source exposed, and toured the RP Calibration Lab. The inspector noted that the source was not located on a workbench, as discussed in the VY evaluation. The source was left exposed on lead block shields at a height of approximately three feet. In addition, the inspector determined that the source was left exposed for several hours and an individual was in the Calibration Lab during that time, unaware of the potential danger created by the exposed source.

The inspector concluded that the VY event evaluation was not sufficiently detailed and that additional evaluation of this event was warranted. This issue will be reviewed further during routine NRC review of the VY Radiological Protection Program.

3.2.2 **Radiation Protection Program Changes**

A Radiation Protection Program Improvement Plan was approved for implementation by the Plant Manager on May 15. This program covers such areas and issues as: controls over high radiation access, communications to workers, methods for contaminated area setup and control, control of radiation protection equipment, RP department practices and staffing levels, training, Health Physics Information System use, contamination event reporting, plant decontamination efforts, and laundry and trash handling. The plan includes scheduling milestones for each of the plan areas.

On May 13, VY announced that the plant health physicist will serve an approximately two-year assignment as the RP Training Coordinator in the training department. This selection is part of VY's Rotation Development Program for the RP department staff.

The VY efforts to improve performance in this functional area will be reviewed on an ongoing basis by the NRC.

4.0 **MAINTENANCE AND SURVEILLANCE (62703, 61726, 71707)**

4.1 **Maintenance Inspection Activity**

The inspector observed selected maintenance activities on safety related equipment to ascertain that these activities were conducted in accordance with approved procedures, TS, and appropriate industry codes and standards.

4.2 **Maintenance Observations**

4.2.1 **LCO Maintenance, Diesel Fire Pump**

On April 16, the diesel driven fire pump (P40-1A) was removed from service for preventive maintenance. The diesel driven fire pump (DDFP) was scheduled to be out-of-service for 18 days. During this preventive maintenance period the DDFP was disassembled, inspected, and refurbished. The appropriateness of the LCO-maintenance activity on the DDFP was reviewed and approved by the PORC. The inspector subsequently concluded that the "at power" inspection of the DDFP provided a net safety benefit to the plant.

According to the VY Maintenance Department, DDFP vendor information contains no overhaul recommendations. However, based on results from service water pump inspections, VY considered inspection of the DDFP a prudent action. The DDFP has been in service for approximately 20 years and a detailed inspection/overhaul had not been completed.

When the DDFP was disassembled, maintenance personnel noted corrosion/erosion of the grey cast iron pump bowl, flow diffusers and the suction manifold. VY determined that the corrosion phenomenon, known as graphitization, resulted in a maximum pressure boundary metal loss of approximately 3/16 of an inch. The pump bowls are designed with a safety factor of 5 with respect to the pressure boundary. According to a pump vendor (Peerless) representative, the conditions found during the DDFP inspection would not have resulted in an instantaneous failure of the pump. Normal DDFP parameter monitoring (vibration, flow, discharge pressure) would have detected this degradation prior to an instantaneous failure of the pump.

The pump bowl was replaced and degraded components replaced or refurbished. After satisfactory post-maintenance testing, the DDFP was declared operable on May 3.

On May 23, in accordance with VY Technical Specification (TS) Section 3.13.B.2, VY submitted a letter to the NRC (BVY 91-55) notifying the NRC that a Vital Fire Suppression Water System component was inoperable for greater than seven days and outlining the plans and procedures to be used to compensate for the loss of system redundancy.

Based on the results of the DDFP inspection and the similarity in pump design, the Maintenance Department developed a Justification for Continued Operation (JCO) for the electric motor driven fire pump. The JCO considered maintenance history, surveillance history, metallurgy and pump design. The JCO concluded that the pump design was substantial and based on the level of deterioration found in the DDFP, there were reasonable assurances that the electric fire pump would remain operable. An overhaul of the electric fire pump is scheduled for September 1991.

4.2.2 Standby Liquid Control Pump Maintenance (UNR 91-12-03)

During the period of May 14-16, VY conducted maintenance activities on the Standby Liquid Control (SLC) pumps P45-1A and -1B. TS 3.4.B Limiting Condition for Operation allows one of the redundant pumps to be out-of-service for a period of seven days. Two different maintenance issues were being addressed by the conduct of this maintenance. The first issue involved the Operations Department identifying on October 31, 1990 that the P45-1A and -1B pumps had packing leakages of 62 and 14 drops per minute leakage, respectively. Corrective Maintenance Requests (CMR) No. 90-3226 and 90-3217 were issued to identify and initiate correction of this condition. The second issue involved the periodic replacement of motor and pump mounting bolts, as documented on CMR Nos. 90-3132 and 90-3133.

Regarding the corrective maintenance conducted to address the packing leakage, the inspector noted that an equipment failure and probable cause review was conducted in accordance with the root cause failure evaluation process described in procedure AP 0200,"Conduct of Maintenance Activities." This review documented that the probable root cause of the condition was that the pumps were operated without coolant The recommendations to prevent recurrence specified that the pump is to be run with coolant. The Maintenance Department determined that during a previous monthly Inservice Test, an Auxiliary Operator (AO) failed to ensure that the SLC test tank water level was above the minimum water level mark, as prescribed in procedure OP 4114,

"Standby Liquid Control System Surveillance." Although no specific corrective action record was generated, the inspector was informed by the Senior Operations Engineer that the Operations Planning Coordinator, during his procedurally required review of the completed maintenance, brought the issue to the attention of the Operations Supervisor. Subsequently, the individual Shift Supervisors were directed to address this attention-to-detail/procedural adherence issue with the shift AOs.

A second issue identified by the inspector involved the acceptance criteria used by the Maintenance Department during their post-maintenance testing. MR 90-3226 for SLC pump P45-1A specified three drops per minute maximum leakage, whereas the stated acceptance criterion for the SLC pump P45-1B specified in MR 90-3217 was no abnormal (excessive) packing leakage. Although the latter criterion is contained in the Vermont Yankee Equipment Manual No. 0088, the inspector noted that the procedure OP 4114 stated that the packing leakage should be between 0-10 drops per minute at each plunger.

Further inspector review of Maintenance Department records involving the replacement of motor and pump mounting bolts was conducted. The inspector reviewed MR 91-0868 issued on April 17 to replace a bent motor mounting bolt in the SLC pump P45-1B. Also, Nonconformance Report (NCR) No. 91-003 was issued to address deficiencies involving a preventive maintenance inspection of SLC pump mounting bolts which identified rejectable material during visual examination; however, the pump was placed back in service without resolution of the identified defect. MR 91-0868 was the immediate corrective action for this identified deficiency. The NRC inspector noted that this issue and its root cause is similar to a prior discrepancy and corrective action documented in Audit Report No. VY-90-17 and Corrective Action Report (CAR) No. 91-11.

VY's actions to remove the SLC pumps from service to conduct corrective maintenance was appropriate. Proper planning and concern for safety related maintenance back-log was evident. Equipment out-of-service time was minimized, root cause evaluations were conducted, actions were taken to prevent recurrence, and post-maintenance testing was accomplished. The establishment of a universally used acceptance criteria for packing leakage, as well as documentation of this criteria in the applicable equipment manual is warranted. The adequacy of the corrective actions associated with recurring deficiencies in the visual examination process, as documented in NCR 91-003 and CAR No. 91-11, remains an unresolved item (UNR 50-271/91-12-03).

4.3 Surveillance Inspection Activity

The inspector performed detailed procedure reviews, witnessed in-progress surveillance testing, and reviewed completed surveillance packages. The inspector verified that the surveillance tests were performed in accordance with TSs, approved procedures, and NRC regulations.

4.4 Surveillance Observations

4.4.1 Routine Observations

The inspector reviewed, and observed selected portions of the following surveillances:

-- AP 0206, Rev. 2, "Inservice Testing Vibration Measurements."

-- OP 4376, Rev. 14, "Torus-Reactor Building Vacuum Breaker Differential Pressure Functional/Calibration."

-- OP 4506, Rev. 1, "Operation and Source Calibration of Reactor Building Ventilation Exhaust Air, AOG Building Ventilation Exhaust Air and Containment Air Monitors."

-- OP 4114, Rev. 24, "Standby Liquid Control System Surveillance."

-- OP 4605, Rev. 19, "Environmental Radiation Sampling and Analysis."

Based upon a review of activities in this area, the inspector noted the following.

-- On April 16, during the performance of OP 4605, VY discovered that power to the Northfield, MA air sampling station (No. 14) had been interrupted for half the bi-monthly sampling process. Potential Reportable Occurrence No. 91-28 was developed to document this inadvertent missed surveillance. This event was determined to not be reportable due to the note contained in TS Table 3.9.3 that deviations are permitted from the required sampling schedule if specimens are unobtainable due to malfunctioning equipment. VY plans to report this event in the annual Radiological Environmental Surveillance Report, as required by this TS.

-- On April 23, during the performance of procedure OP 4376, Instrument and Control Department technicians noted that the differential pressure indicating switch, DPIS 16-19-32B, setpoint slightly exceeded the less than or equal to 0.5 psid TS 3.7.A.5, a required setpoint. This instrument is one of the two redundant devices used to operate the pressure suppression chamber to reactor building vacuum breakers. Corrective actions to correct the deficiency were implemented and a determination of event reportability in accordance with VY procedure OP 0010 was conducted.

The inspector concluded that the observed surveillance testing effectively met the safety objectives of the surveillance testing program.

5.0 SECURITY (71707, 93702, 92700)

5.1 Observations of Physical Security

Compliance with the security program was verified on a periodic basis, including the adequacy of staffing, entry control, alarm stations, and physical boundaries.

5.2 Non-Specific Threat

On May 7 at 7:10 p.m., a non-specific telephone threat was received by the Access Control Officer at Gatehouse No. 2. Notification of the event was made to the Shift Supervisor and cognizant plant management. Although the event did not appear to represent a credible threat, prudent compensatory measures were implemented and appropriate law enforcement agencies notified. The inspector, NRC Operations Center, and the cognizant NRC Region I (NRC:RI) security specialist inspector were notified on an informational basis.

VY response to this event was characteristic of a conservative and professional approach. Good management involvement and timely communications between on-site personnel, VY security management, and off-site law enforcement agencies was noted.

5.3 Relocation to Gatehouse No. 3

During this inspection period, VY continued construction activities on its Gatehouse No. 2 upgrading project. This gatehouse is the normally used main personnel entranceway into the Protected Area (PA). For an approximate two month period starting on June 3, personnel access into the PA will be accomplished through Gatehouse No. 3. The intended change to site operations and interim PA perimeter configuration was described in VY letter No. 91-130 to NRC:RI transmitted on May 28.

On May 31, the inspector observed VY security supervision and contractor security personnel evaluating the interim access control features of Gatehouse No. 3. Equipment currently in Gatehouse No. 2 that is required to support the VY Emergency Plan and Implementing Procedures was scheduled for relocation to Gatehouse No. 3 on June 3. Detailed equipment and process checkouts were conducted. A good level of security management involvement and oversight was exercised. Consideration for proper fire protection in the vicinity of the wood cooling towers was evident by the prohibition of parking of vehicles near the towers and is indicative of the high level of preplanning associated with the interim changes necessary in the area of access control into the PA.

5.4 Anti-Nuclear Demonstration

On April 27, a demonstration occurred at 4:00 p.m. at the main gate of the VY owner controlled area (OCA) by approximately 45 members of anti-nuclear citizen groups. Five of the demonstrators were taken into custody by the local law enforcement agencies for disorderly

conduct and obstructing traffic. One demonstrator managed to enter the area between the two vehicle gates and was confronted within the OCA by VY security personnel. The inspector observed the demonstration and VY security organization activities from Gatehouse No. 1. Demonstrators that were confronted by VY security personnel or local authorities were taken into custody without active resistance and appeared cooperative with authorities. The demonstration ended at approximately 6:00 p.m. on April 27.

VY response to this event was noteworthy. The security staff performed in a professional manner and displayed restraint and respect in dealing with the demonstrators. VY briefed the inspector on their contingency planning for the announced demonstration, and provided evidence that they had a good understanding of the situation and the need to prioritize the security response to focus on the PA. Good cooperation and communications were noted between security and operational personnel. An appropriate review of the event for security notifications was made. The communications and coordination between the VY security and local authorities was excellent, and reflected a high degree of preparation. No challenges to the plant PA occurred and no impact on plant operations occurred as a result of this demonstration.

6.0 ENGINEERING AND TECHNICAL SUPPORT (71707, 37700, 92700)

6.1 Environmental Qualification Concerns Due to Unanalyzed Heating Steam Line (LER 91-08)

On March 25, VY determined that failure of non-seismic qualified piping in the house heating steam lines had not been considered in previous high energy line break (HELB) analyses. The failure of house heating steam lines could potentially expose safety-related electrical equipment to environmental conditions outside those evaluated in the VY Environmental Qualification (EQ) program. Additional, information concerning this event is contained in NRC Inspection Report 50-271/91-11, Section 6.3.

On May 1, VY reported this event to the NRC in Licensee Event Report (LER) 91-08, "Potential Environmental Conditions Not Previously Evaluated as a Result of Omission from Original Line Break Analysis."

VY determined the root cause to be personnel error in that the house heating system was omitted from HELB analyses performed in 1973 and 1974. VY also determined that the root cause of the failure to provide a timely assessment of NRC Information Notice 90-53 was attributed to an incomplete procedure. Specifically Procedure AP 0028, "Operating Experience Review and Assessment/Commitment Tracking" does not provide adequate guidance as to what actions should be taken if the item potentially impacts plant operations or safety.

A Corrective Action Plan continued to be developed and additional corrective actions associated with this event were targeted for completion by June 21. The inspector concluded that corrective actions described in LER 91-08 were appropriate.

6.2 Feedwater Check Valve Flaws

In 1989 and 1990 VY identified cracks in the stellite wear pads in the piston guide portion of the feedwater check valves. This issue and VY corrective actions were discussed in NRC Inspection Reports 50-271/89-02, 90-15, and 90-17. There are three feedwater check valves in series in each of the two feedwater lines. Four of the six lift check valves have already been replaced with swing check valves. During the next refueling outage, the remaining two valves (27B and 96B) will also be replaced with swing check valves.

One of the lift check valves (28A) removed from service during the 1990 refueling outage was sent by VY to the Brookhaven National Laboratory (BNL) for metallurgical characterization. This action was in support of NRC research efforts to conduct failure analysis on nuclear equipment. On May 7 an NRC:RI specialist inspector conducted an inspection at BNL to review the program relating to the characterization of flaws found in the stellite wear strips of the subject feedwater check valve.

The inspector reviewed the history of the feedwater check valve stellite wear strip cracking, VY inservice inspection data, check valve maintenance records, prior inspection, and the program of flaw characterization being implemented at BNL.

BNL has taken trepanned samples containing flaws from the check valve body to provide for a sample source from which the characterization of the flaws may be determined by metallographic and fractographic analysis. On completion of the studies, a technical report will be written by BNL. On completion of this project, the check valve will be cleaned and disposed of using BNL equipment and facilities capable of handling contaminated material.

Although the characterization study by BNL was not completed at the time of the inspection, the inspector noted that the cracks were located in the stellite overlay regions at the inner diameter surface of the 28A valve body. Ultrasonic testing of the valve body prior to disassembly had indicated cracks in a through-wall direction of 0.4 to 1.1 inches deep in wall thicknesses of 2.05 to 2.45 inches, respectively. The cracks appeared to be linear tears in the stellite, some of which extended through the stellite/base material interface. Samples of the trepanned check valve body contained flaws of varying size and orientation and some cracks appeared not to project into the valve body base material. The stellite is grade No. 6, but the process utilized to deposit the stellite was not known by BNL and could not be determined from the data reviewed.

The inspector had no further questions on this matter at this time.

6.3 Emergency Response Facility Information System Enhancement

The Emergency Response Facility Information System (ERFIS) is a process computer system at VY that provides for a variety of functions. These functions are the safety parameter display system (SPDS), sequence of events post-trip logs, traversing in-core probe system flux data acquisition, rod worth minimizer system, control rod monitoring and analog and digital data

acquisition from balance of plant and nuclear steam supply systems. On May 9, VY implemented a hardware upgrade that increased available memory by 100 percent and central processing unit processing capability by 50 percent. This enhancement to the system will shorten system response time for the various ERFIS programs. Plant Operators, who utilize the ERFIS color graphic displays for both routine and off-normal plant operations, will observe improved information display response time.

To accomplish the above enhancements, VY removed the ERFIS from operation for an eight hour and five minute period on May 9. With the system out-of-service, the SPDS function was rendered inoperable. In accordance with VY reportability procedure requirements, a report to the NRC was made to the NRC Operations Center due to the loss of emergency assessment capability [10 CFR 50.72(b)(1)(v)].

VY's reportability determination was appropriate and their actions to provide important enhancements to the ERFIS system reflect a performance strength in this functional area.

7.0 SAFETY ASSESSMENT AND QUALITY VERIFICATION (90713, 90712, 92700)

7.1 Licensee Event Reports (LERs)

The inspector reviewed the LERs listed below and determined that, with respect to the general aspects of the events: (1) the report was submitted in a timely manner, (2) the description of the event was accurate, (3) a root cause analysis was performed, (4) safety implications were considered, and (5) corrective actions implemented or planned were sufficient to preclude recurrence of a similar event.

LER 91-08, "Environmental Conditions Not Previously Evaluated as a Result of Omission From Original Line Break Analyses," (Section 6.1).

LER 91-10, "Failed Relay Coil Results in Primary Containment Isolation System Actuation," (Section 2.2.5).

7.2 Periodic and Special Reports

The plant submitted the following periodic and special reports which were reviewed for accuracy and the adequacy of the evaluation:

-- Monthly Statistical Report for April.

-- Vermont Yankee Annual Radiological Environmental Surveillance Report for calendar year 1990.

These reports were found to be timely and adequate in scope and content.

8.0 MANAGEMENT MEETINGS (30702)

8.1 Preliminary Inspection Findings

At periodic intervals during this inspection, meetings were held with senior plant management to discuss preliminary inspection findings. A summary of findings for the report period was also discussed at the conclusion of the inspection and prior to report issuance. No proprietary information was identified as being included in the report.

An unresolved item is a matter about which more information is required to ascertain whether it is an acceptable item, a deviation or a violation. Unresolved items are discussed in Sections 2.2.7 and 4.2.2 of this report.

8.2 Region Based Inspection Findings

Two Region based inspections were conducted during this inspection period. Inspection findings were discussed with senior plant management at the conclusion of the inspections.

Date	Subject	Rpt. #	Inspector
4/25-29/91	AIT Review of Loss of Off-Site Power Event	91-13	C. Anderson
5/14-15/91	Operator Licensing	91-15	J. Williams
5/20-24/91	Team Inspection: MOV Program Review (TI 2515/109)	91-80	J. Yerokun

8.3 Site Visits and Significant Meetings

-- An NRC Region I Systematic Assessment of Licensee Performance (SALP) Board met on April 30 at the NRC:RI office to review and evaluate the performance of activities associated with station operation for the period of October 1, 1989 through March 16, 1991. The initial SALP Report, 50/271/89-99, was issued on May 20.

-- On May 7, the Directors of NRC:RI Division of Reactor Projects and NRC Office of Nuclear Reactor Regulation Project Directorate I-3 toured VY facilities and reviewed ongoing activities.

-- On May 14, an exit meeting was held at the NRC:RI office to review the results of the AIT 50-271/91-13.

-- On May 23, a member of the NRC Advisory Committee on Reactor Safeguards toured VY facilities and met with senior plant and corporate management.

-- U.S. Representative Peter Kostmayer and his Congressional staff visited the VY site on May 27. In addition to the site tour and briefing provided by VY, the NRC:RI Deputy Regional Administrator and resident inspectors provided a briefing on NRC activities involving VY.