January 18, 2000

EA 99-299

Mr. Samuel L. Newton
Vice President, Operations
Vermont Yankee Nuclear Power Corporation
185 Old Ferry Road
Brattleboro, Vermont 05301

SUBJECT: NRC INTEGRATED INSPECTION REPORT NO. 05000271/1999009

Dear Mr. Newton:

On December 5, 1999, the NRC completed an inspection at your Vermont Yankee (VY) facility. The enclosed report presents the results of that inspection.

During the six weeks covered by this inspection, the conduct of activities at Vermont Yankee was characterized by a safe plant shutdown and a well controlled maintenance outage. Your efforts to improve outage planning and implementation were evident in the execution and control of in-plant work. Good performance was noted during our reviews of radiological controls, engineering support, and inservice inspection activities. Although four violations of NRC requirements were identified during this period, only the violation associated with radiological controls occurred during the outage.

Based on the results of this inspection, the NRC has determined that four Severity Level IV violations of NRC requirements occurred. These violations are being treated as Non-Cited Violations consistent with Section VII.B.1.a of the Enforcement Policy (NUREG 1600, November 9, 1999). The NCVs involved (1) a failure to translate RHR valve design information into operating procedures, (2) a failure to perform a safety evaluation during a 1990 EOP change, (3) an incomplete design review to change an isolation valve's normal position, and (4) a failure to adequately inform workers of radiation dose rates. The NCVs are further described in the subject inspection report. If you contest the violation or severity level of the NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region I; and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Vermont Yankee facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practices," a copy of this letter and its enclosure will be placed in the NRC Public Document Room.

Sincerely,

/RA/

Clifford J. Anderson, Chief
Projects Branch 5
Division of Reactor Projects

EA 99-299
Docket No.: 05000-271
License No.: DPR-28

Enclosure: NRC Inspection Report 05000271/1999009

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U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.	05000271
Licensee No.	DPR-28
Report No.	05000271/1999009
Licensee:	Vermont Yankee Nuclear Power Corporation
Facility:	Vermont Yankee Nuclear Power Station
Location:	546 Governor Hunt Road Vernon, Vermont
Dates:	October 25 - December 5, 1999
Inspectors:	Brian J. McDermott, Senior Resident Inspector Edward C. Knutson, Resident Inspector Ronald L. Nimitz, Senior Radiation Specialist Raymond K. Lorson, Senior Resident Inspector - Seabrook Javier M. Brand, Resident Inspector - Seabrook Thomas F. Burns, Reactor Engineer Barry S. Norris, Senior Reactor Inspector
Approved by:	Clifford J. Anderson, Chief Projects Branch 5 Division of Reactor Projects

EXECUTIVE SUMMARY

Vermont Yankee Nuclear Power Station
NRC Inspection Report 05000271/1999009

This integrated inspection included aspects of licensee operations, maintenance, engineering, and plant support. The report covers a six week period of routine activities by the resident inspectors assigned to VY and resident inspectors for another Region I reactor facility. In addition, inspectors from the NRC Region I office conducted reviews of radiological protection, inservice inspection, and engineering support.

Operations

- Plant operations were well controlled during the shutdown to begin the 1999 refueling outage. Several equipment problems were appropriately addressed and had no significant operational effect. Unnecessary activities in the control room were minimized and operators were appropriately focused on reactivity manipulations. (Section O1.2)

- A planned swap of shutdown cooling subsystems shortly after the reactor shutdown was appropriately evaluated for risk considerations. Primary containment was maintained as a conservative compensatory measure during the swap. The operation was completed with no problems. (Section O1.3)

- The licensee responded well to a mis-positioned fuel assembly within the spent fuel pool. This event did not adversely affect plant safety. (Section O4.1)

- VY's procedure for the control of staff working hours was consistent with regulatory guidance. An audit of selected staff and contractor working hours during the 1999 refueling outage found that the procedure controls were effective. No operational or performance events were found attributable to the excessive use of overtime. (Section O8.1)

Maintenance

- Planned maintenance and surveillance activities observed during the 1999 Refueling Outage were well controlled and executed. VY management's emphasis on in-field oversight was apparent in that good supervisory oversight and support were noted during NRC observations of significant work activities such as the DC-2 bus maintenance. (Section M1.1)

- The VY organization responded well when a replacement control rod drive mechanism became stuck as operators attempted to couple its control blade. There was no immediate safety issue associated with the stuck mechanism and an appropriate safety focus was maintained during the development and execution of a recovery plan. VY completed appropriate inspections of the control blade prior to the installation of a new drive mechanism. (Section M1.2)

- The source range monitoring system neutron flux trip setpoints were properly reduced to support the plant refueling operations. The inspector observed some work performance

deficiencies during this activity, however there was no impact on the final outcome of the task. The licensee's planned and completed actions for addressing the deficiencies appeared appropriate. (Section M1.3)

- VY's initial response and evaluation of a Residual Heat Removal valve performance anomaly was not thorough and did not have adequate engineering involvement. While following up on NRC questions, engineering personnel identified that the initial evaluation was incorrect and that the valve was degraded. When valve RHR-65B was disassembled, VY discovered its stem was broken. (Section M2.1)

- A valve stem failure for one RHR heat exchanger bypass valve (RHR-65B) and a crack in the redundant train's valve stem (RHR-65A) were discovered during the 1999 refueling outage. Pending the results of the licensee's root cause investigation and metallurgical evaluation, this issue will be tracked as an inspector follow-up item. (Section M2.1)

- The design capability of the RHR heat exchanger bypass valves (based on torque switch settings) was not appropriately incorporated into two operating procedures as required by Quality Assurance requirements for Design Control (10CFR50, Appendix B, Criterion III). This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy, (NUREG 1600, November 9, 1999). This violation was entered in VY's corrective action program as ER 99-1493 and ER 99-1427. (Section M2.1)

- Inservice inspection activities during the 1999 refueling outage were performed acceptably and included acceptable ASME program coverage, qualified personnel, approved procedures, proper implementation, appropriate examination documentation, and VY oversight. NDE results were well documented and indications were appropriately recorded and resolved. VY's inspections were thorough and of sufficient extent to determine the integrity of the components inspected. (Section M2.2)

- Overall, the control of contractors and station personnel with regards to following procedures has improved since the last refueling outage. The additional oversight helped to ensure the proper identification and correction of problems. A review of QA observations and Event Reports from the outage found few issues directly or indirectly related to contractor activities. Good oversight was provided by the designated VY representatives and frequent QA observations. (Section M7.1)

Engineering

- An NRC review of Emergency Operating Procedure actions for containment flooding identified that VY's 1990 incorporation of generic guidance resulted in an unreviewed safety question (USQ). VY's implementation of this change, without prior NRC approval, is a violation of 10CFR50.59. The procedures have since been appropriately revised and this error is not considered indicative of current licensee performance. Due to the overall low risk significance of containment flooding, this violation is being treated as a Severity Level IV, Non-Cited Violation, consistent with Section VII.B.1.a of the NRC

Enforcement Policy, (NUREG 1600, November 9, 1999). This violation was entered in VY's corrective action program as ERs 97-0273, 97-0479, and 97-1328.

- VY's failure to evaluate the effect of containment flooding during a 1992 design change for a containment hardened vent is a violation of Quality Assurance requirements (10CFR50 Appendix B, Criterion III, Design Control). Potential problems were resolved by subsequent procedure changes and this error is not indicative of current engineering performance. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy, (NUREG 1600, November 9, 1999). This violation was entered in VY's corrective action program as ER 97-306. (Section E8.1)

Plant Support

- VY continued to maintain a good radiation protection program. In general, improved work planning and controls were well implemented. Emergent work was formally evaluated and approved for addition to outage scope. No changes were identified that adversely affected radiation protection program performance. (Section R1.1)

- Overall, VY implemented an effective ALARA program. There was effective planning and preparation for outage radiological work activities. VY implemented good efforts to reduce personnel occupational exposure for work activities to as low as is reasonably achievable. (Section R1.2)

- One violation for the failure to inform workers of elevated radiation dose rates in their work area was identified by the NRC. This Severity Level IV violation is being treated as an NCV, consistent with Section VII.B.1.a of the NRC Enforcement Policy, (NUREG 1600, November 9, 1999). The violation was entered in VY's corrective action program as ER 97-1503. (Section R1.2)

- Applied radiological controls for ongoing work activities were generally well implemented. No significant unplanned personnel external or internal exposures were identified. No significant airborne radioactivity was identified and no individuals sustained any significant airborne radioactivity intake. (Section R1.3)

- Overall, VY implemented a good radioactive material and contamination control program. Radioactive material was properly labeled, stored, and controlled; contamination monitoring equipment was operable, within calibration, and properly used by personnel. Radiation and contamination surveys were observed to be generally comprehensive and detailed with some exceptions noted. There were minimal instances of personnel contamination during the outage and no significant dose consequences. (Section R1.4)

- VY implemented overall good oversight of ongoing radiological controls activities. There was good in-field presence by QA personnel and supervisors were conducting daily observations of ongoing work activities. Performance assessment findings were provided to plant management. (Section R7.1)

Executive Summary (cont'd)

- VY implemented overall good housekeeping within radiological controlled areas. (Section R8.1)

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Report Details

Summary of Plant Status

At the beginning of the inspection period, the Vermont Yankee (VY) plant was preparing for its twenty first refueling outage. On October 29, operators reduced reactor power and at 12:15 p.m. they disconnected the main generator from the grid. The plant had operated for 372 consecutive days prior to the shutdown, a new record for VY. In addition to refueling the reactor for its next 18 month operating cycle, VY personnel and contractors completed a number of maintenance and modification activities during this outage. On December 2, operators synchronized the main generator to the grid, marking the completion of a 34 day outage.

I. Operations

O1 Conduct of Operations[1]

O1.1 Observation of Routine Plant Operations (71707)

The inspectors routinely toured the control room to assess the conduct of activities, verify the alignment of decay heat removal systems, and confirm the availability of standby systems required by the Technical Specifications (TS) or administrative procedures. Equipment deficiencies and issues identified in control room logs were reviewed and discussed with shift supervision, to evaluate both the equipment's condition and the adequacy of VY's initial response to the issue. No problems with the operation of the plant or status of standby systems were identified by the inspectors.

O1.2 Plant Shutdown for Refueling Outage

a. Inspection Scope (71707)

The inspector observed portions of the October 29 reactor shutdown from the control room.

b. Observations and Findings

Power reduction was commenced at 7:00 a.m. Some minor speed instability was noted on the "B" reactor recirculation motor generator early in the power reduction. Personnel were dispatched to check the motor generator, and no obvious problems were identified. The condition was evaluated as not presenting an immediate operational concern, and was closely monitored during subsequent power reduction with recirculation flow. The inspector concluded that this issue was appropriately dealt with by the control room operators.

During the control rod insertion for power reduction, a problem developed with the rod worth minimizer. When attempts to correct the problem were unsuccessful, the rod worth minimizer was declared inoperable. Power reduction with control rods was

[1]Topical headings such as O1, M8, etc., are used in accordance with the NRC standardized reactor inspection report outline. Individual reports are not expected to address all outline topics.

resumed using two operators to verify proper control rod selection, as allowed by the TS. The inspector observed that operations department management was consulted in developing this course of action, and concluded that the issue was appropriately resolved.

The inspector observed control room operations during the reactor shutdown. The operators that were inserting control rods were appropriately focused on the task. Work control center personnel outside of the control room handled routine plant maintenance activities and thereby helped to minimize distraction of control room personnel. The inspector concluded that plant operations were well controlled during the reactor shutdown.

c. Conclusions

Plant operations were well controlled during the shutdown to begin the 1999 refueling outage. Several equipment problems were appropriately addressed and had no significant operational effect. Unnecessary activities in the control room were minimized and operators were appropriately focused on reactivity manipulations.

O1.3 Shutdown Cooling Operations

a. Inspection Scope (71707)

The inspector observed a transfer of shutdown cooling from the "A" to the "B" subsystem of the residual heat removal (RHR) system.

b. Observations and Findings

VY conducted RHR heat exchanger thermal performance testing on October 30, to make use of the decay heat load available immediately following the shutdown and improve the test accuracy. However, testing both heat exchangers required that shutdown cooling be secured while shifting from one RHR subsystem to the other. The large decay heat load made this a time sensitive operation, since the time for the reactor coolant to boil was approximately two hours and boiling would constitute an inadvertent mode change. VY recognized the risk potential of this evolution and compensated by maintaining primary containment integrity until after shutdown cooling alterations were completed.

Shutdown cooling on the "A" RHR subsystem was secured at 9:15 a.m. The inspector observed operations in the control room to switch cooling to the "B" RHR subsystem, and noted that the activity was well controlled. Shutdown cooling was restored after approximately 50 minutes and reactor coolant temperature increased by approximately 40°F during this time. The coolant heatup was consistent with the predicted heatup rate.

c. Conclusions

A planned swap of shutdown cooling subsystems shortly after the reactor shutdown was appropriately evaluated for risk considerations. Primary containment was maintained as a conservative compensatory measure during the swap. The operation was completed with no problems.

O4 Operator Knowledge and Performance

O4.1 Mis-positioned Fuel Assembly During Core Alteration Activities

a. Inspection Scope (71707)

The inspector reviewed the licensee's response to a fuel assembly that operators moved into the incorrect spent fuel pool storage location during the refueling activities.

b. Observations and Findings

On November 5, the refueling senior reactor operator (SRO) identified that a spent fuel assembly had been improperly transferred from the core into spent fuel pool (SFP) location H41 rather than the intended location at J41. This error was identified at Step 75 of the Core Shuffle 1 procedure when the operators attempted to transfer the assembly from SFP location J41 (which was empty) back into the reactor. The inspector observed that the refueling SRO promptly suspended the refueling activities and contacted the control room. The licensee's immediate corrective actions included:

- The operators were relieved, an event report was initiated, and plant management was briefed on the event.
- The licensee verified (by observation of the bundle serial number), that the assembly in SFP location H41 was actually the bundle that was intended to be placed into position J41. The inspector independently confirmed this observation.
- The next crew of fuel handling personnel were coached regarding the need to perform the fuel movements in strict accordance with the core shuffle procedure.
- The licensee revised the core shuffle procedure to address movement of the fuel assembly from the H41 position.

The inspector concluded that the mis-positioned fuel assembly within the spent fuel pool did not adversely affect plant safety and considered the licensee's actions to be reasonable and complete.

c. Conclusions

The licensee responded well to a mis-positioned fuel assembly within the spent fuel pool. This event did not adversely affect plant safety.

O8 Miscellaneous Operations Issues

O8.1 Overtime Use and Approval

a. Inspection Scope (71707)

The inspector reviewed VY's procedure for controlling the use of overtime to verify that it was consistent with regulatory guidance, and to assess its implementation during the refueling outage.

b. Observations and Findings

The inspector verified that VY administrative procedure (AP) 0894, "Staffing and Overtime Limits," provides adequate controls on the use of overtime that are consistent with Generic Letter (GL) 82-12, "Nuclear Power Plant Staff Working Hours."

The inspector audited the working hours of five operators for a four week period leading up to and during the outage. On one occasion, an operator exceeded the maximum number of hours normally allowed in a one week period and this was authorized in advance as required by AP 0894. The inspector concluded that, within the group that was sampled, the working hours of operations department personnel were being appropriately controlled during the outage.

The inspector audited the working hours of five contract personnel that worked on the main steam isolation valve repairs during the outage. The inspector noted that VY's documentation does not differentiate between work hours and shift turnover time. This is relevant because turnover time is not considered part of the time counted toward the limitations in AP 0894 or GL 82-12. The VY supervisor is responsible for initiating overtime requests in accordance with AP 0894 and several examples were noted where this process had been used. In cases where no requests for additional overtime had been used, the turnover time implied by the contractors' timesheets was not excessive and no violations of AP 0894 were identified. The lack of good documentation to differentiate between work hours and shift turnover time for contract personnel was discussed with the Maintenance Superintendent.

The inspector reviewed VY's corrective action program for issues relating to the use of overtime. Two such event reports had been generated during the outage, but the inspector determined that both issues were administrative in nature and did not constitute significant or programmatic problems. In addition, the inspector questioned VY whether any outage ERs had been determined to have been caused by personnel being overly tired. VY indicated that no such cases had been identified. Furthermore, through periodic attendance of ER screening meetings and monitoring of plant status during the outage, the inspector knew of no operational or performance issues that were related to personnel being overly tired. The inspector concluded that AP 0894 was being appropriately administered, and that the use of overtime during the outage was not contributing to operational or performance related events.

c. Conclusions

VY's procedure for the control of staff working hours was consistent with regulatory guidance. An audit of selected staff and contractor working hours during the 1999 refueling outage found that the procedure controls were effective. No operational or performance events were found attributable to the excessive use of overtime.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Maintenance and Surveillance Observations

a. Inspection Scope (61726, 62707)

The inspector observed portions of plant maintenance activities to verify the use of approved procedures, appropriate conduct and control of the work, and compliance with regulatory requirements. The inspector also observed portions of surveillance activities to verify proper calibration of test instrumentation, use of approved procedures, conformance to Limiting Conditions for Operations (LCOs). Following the completion of maintenance and surveillance activities, the inspector verified that safety systems were returned to their appropriate standby alignments.

b. Observations and Findings

The inspector observed portions of the in-plant work and reviewed work documents associated with the following activities:

DC-2 Bus Preventive Maintenance
The inspector observed various portions of the equipment tag-out, the bus inspection, and breaker replacement activities. The work was performed in accordance with OP 5286, "125V DC-2 Distribution Panel Inspection and Testing" and work order (WO) 99-00217-00. No problems occurred during the de-energization of this essential distribution panel, the breaker replacements and inspections observed were carefully performed, and good supervisory oversight and support was apparent. The maintenance was well planned and executed, and addressed open action items in VY's Maintenance Rule Program performance improvement plan for this risk significant a(1) system.

Control Rod Drive Mechanism (CRDM) Change-outs
On October 31, the inspector observed portions of the CRDM change-out activities. A total of 12 CRDMs were scheduled to be replaced in accordance with OP 5211, "CRDM Removal and Installation Procedure." The maintenance supervisor conducted an excellent pre-job briefing which included a discussion of the procedure and previous experiences with the evolution. Required precautions, and contingencies to prevent or mitigate excessive reactor coolant leakage during a CRDM change out were discussed. The inspector noted adequate field coverage by operations and health physics personnel during the replacement of two CRDMs. However, later in the replacement activity, CRDM 26-03 became stuck during the re-coupling of its control blade. This problem is further discussed in Section M1.2.

Main Steam Isolation Valve Maintenance
Refurbishment of the four inboard main steam isolation valves was one of the longest duration planned activities of the outage. The inspector observed various aspects of this work during routine plant tours and noted no problems.

Reactor Recirculation Pump Seal Replacement
Seal replacement was performed on both of the reactor recirculation pumps as preventive maintenance. The inspector observed various aspects of this work during routine plant tours and noted no problems.

Main Station Battery Service Test
The inspector observed preparations for the discharge test of the B-1-1B Main Station Battery, reviewed OP 4215, "Main Station Battery Performance/Surveillance Test," and discussed the test with cognizant VY personnel. The inspector noted that VY changed the sequence of work on the battery system so that the post test recharge could be performed using the battery charger normally aligned to bus DC-2. This substantially reduced the potential for a repeat of the battery shorting event which occurred during the 1998 refueling outage (reference NRC Inspection Report 50-271/98-04). The activity was well controlled, the battery performance was acceptable, and no problems were identified.

Motor-Operated Valve Dynamic Testing
Dynamic testing of the turbine building service water isolation valves, V70-19A and V70-20, was performed on November 20 using a special test procedure, STP 99-002. The test was performed as a Special Test Procedure because of the potential for water hammer when restoring service water system flow. A licensed senior reactor operator acted as a dedicated test coordinator. Testing was completed satisfactorily and the inspector noted no evidence of water hammer during a post-test examination of susceptible service water system components.

Emergency Core Cooling System Integrated Automatic Initiation Test
This test was performed on November 24 in accordance with OP 4100, revision 26. Along with satisfying TS surveillance requirements, this test serves as a proof test for the ECCS, in that equipment is actually started and run in the automatic sequence. The smooth performance of this test was a positive reflection on management's oversight of the extensive test preparations, as well as the material readiness of the ECCS.

Standby Liquid Control System Surveillance and Testing of Squib Valves
These activities were performed in accordance with OP 4114 and OP 4203 respectively. The testing was completed satisfactorily and no problems were identified by the inspector.

High Pressure Coolant Injection Turbine Exhaust Line Modifications
The modification was performed in accordance with engineering design change request (EDCR) 98-409. The inspector found the work performed under this activity to be professional and thorough. All work observed was performed with the work package or procedure present and in use. Technicians were knowledgeable of their assigned tasks. The inspector observed appropriate supervisory oversight, and proper industrial safety and radiological work practices.

c. Conclusions

Planned maintenance and surveillance activities observed during the 1999 Refueling Outage were well controlled and executed. VY management's emphasis on in-field oversight was apparent in that good supervisory oversight and support were noted during NRC observations of significant work activities such as the DC-2 bus maintenance.

M1.2 Control Rod Drive Mechanism (CRDM) Binding

a. Inspection Scope (62707)

On November 1, the CRDM for control rod 26-03 was replaced as part of the planned outage maintenance. However, the CRDM became stuck in essentially the fully withdrawn position when operators attempted to perform the post-replacement coupling check. The inspector observed VY's response to this event, including a number of response team meetings and the implementation of their recovery plan.

b. Observations and Findings

VY's initial investigation of the event determined that the control rod and/or the CRDM was stuck between position 47 and 48. After consultation with the CRDM vendor and a knowledgeable contractor, the licensee developed a recovery plan. A revision to the work order for replacement of the CRDM, which was reviewed by VY management, allowed the use of a hydro pump to pressurize the above-piston area to a maximum of 1000 psig to withdraw the mechanism.

On November 2, the hydraulic control unit for CRDM 26-03 was valved out of service, the hydro pump was connected to a withdrawl line vent, and a drain hose was connected to the insert line vent. This configuration simplified the operator actions necessary to slowly increase the above-piston pressure using the hydro pump. At approximately
780 psig, the CRDM withdrew to the overtravel position. After three different methods were used to verify the control blade was properly backseated (providing isolation for the control rod guide tube), maintenance personnel were able to remove the CRDM using normal procedures.

A thumb screw (approximately 1.5 inches in length) was found between the outer filter and the index tube of the CRDM. Apparently, the screw had become dislodged from one local power range monitor (LPRM) cable clamp that was partially disassembled during preparation for the under vessel work. It is likely that the thumb screw fell into

the top of the CRDM as the workers were moving the CRDM to a vertical orientation for installation.

After VY performed inspections to verify there was no damage to the control blade, a new CRDM was installed and tested without incident.

c. Conclusions

The VY organization responded well when a replacement control rod drive mechanism became stuck as operators attempted to couple its control blade. There was no immediate safety issue associated with the stuck mechanism and an appropriate safety focus was maintained during the development and execution of a recovery plan. VY completed appropriate inspections of the control blade prior to the installation of a new drive mechanism.

M1.3 Calibration of Source Range Monitor (SRM) Setpoints

a. Inspection Scope (62707)

The inspector observed an instrument and controls (I&C) technician reduce the SRM high and high-high neutron flux trip setpoints in accordance with procedure, OP 5371, “Instrument and Control Routine Outage Activities.” The SRM setpoints were reduced to conservative administrative limits to support the outage refueling activities.

b. Observations and Findings

The inspector observed the work activities and independently confirmed that:

- The four SRM channel setpoints were reduced to the values specified in the OP 5371 procedure;
- All test equipment was within its calibration periodicity;
- The SRM channels were properly restored to service following the maintenance activities.

The inspector observed some performance deficiencies during this activity, including:

- The technician incorrectly opened the "B" SRM drawer while the "C" SRM drawer was in the bypass condition. This error was promptly identified by another I&C technician at the job site.
- The technician performed a second adjustment of the test equipment output voltage (near the SRM trip setpoint) that was not addressed in the OP 5371 procedure. The inspector discussed this issue with an I&C engineer who indicated that the second voltage adjustment would not change the SRM setpoint. This issue was also discussed with I&C Management who stated that the procedure would be revised to provide better guidance for establishing the test equipment output voltage.

Neither of the above deficiencies resulted in a plant problem, however they highlighted areas for improvement in I&C work performance. The inspector concluded that the licensee's immediate and planned actions discussed above appeared appropriate.

c. Conclusions

The source range monitoring system neutron flux trip setpoints were properly reduced to support the plant refueling operations. The inspector observed some work performance deficiencies during this activity, however there was no impact on the final outcome of the task. The licensee's planned and completed actions for addressing the deficiencies appeared appropriate.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Residual Heat Removal Valve Stem Failure

a. Inspection Scope (62707)

On October 30, the RHR heat exchanger bypass valve (V10-65B) did not fully close when operated from the control room. At the time, operators were preparing for a thermal performance test of the "B" RHR heat exchanger. In order to continue with the test, the valve was manually closed using the motor-operator's handwheel. The inspector reviewed the licensee's response to this anomaly, based on the safety significance of the RHR system and its components.

b. Observations and Findings

Event Report 99-1427 was initiated based on the valve's failure to fully close and an operability determination checklist evaluation concluded the valve could be considered operable, but requested additional engineering evaluation.

On November 2, the inspector reviewed a written evaluation of the valve anomaly prepared by the Work Management Group. The evaluation reasoned that the valve did not fully close because the differential pressure created during the test exceeded the design basis differential pressure that had been used to establish the valve's torque switch settings. VY concluded that the valve was operable because it would not experience a high differential pressure under accident conditions and that the test procedure was the cause of the problem.

The inspector observed that the November 2 evaluation did not specifically discuss how much bypass flow would be acceptable under design conditions or whether other procedures allowed valve alignments that would create a high differential pressure across the heat exchanger bypass valves. These questions were discussed with a supervisor in Design Engineering.

After a subsequent review of the event, VY reached several new conclusions. Most importantly, VY recognized that RHR-65B was not exposed to a high differential pressure during the test and that significant flow (~1000 gpm) through RHR-65B existed

after the operators had manually closed it. In addition, Operations management determined that RHR shutdown cooling procedure OP 2124 did not preclude operation of the RHR system in a way that would cause an excessive differential pressure across the bypass valves.

The inspector concluded that two RHR operating procedures were inconsistent with the design of the valves. Specifically, OP 4032 for heat exchanger thermal performance testing, and OP 2124 for shutdown cooling, did not preclude operation of the heat exchanger bypass valves under dynamic conditions that would exceed their design basis. 10CFR50 Appendix B, Criterion III, Design Control, requires that design information be correctly translated into procedures and instructions. Contrary to the above, the design capability (based on torque switch settings) of the RHR heat exchanger bypass valves was not appropriately incorporated into two operating procedures. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy, (NUREG 1600, November 9, 1999). This violation was entered in VY's corrective action program as ER 99-1493. **(NCV 05000271/1999009-01: Failure to Translate MOV Design into Operating Procedures)**

Once the engineers determined that RHR-65B's operation was not as expected, they initiated a more detailed investigation. ER 99-1471 identified the potential for containment cooling to be effected by the failure of an RHR heat exchanger bypass valve to close. This event report was screened by VY management on November 3.

On November 15, VY performed a check of the motor-operator for RHR-65B and found no problem. However, during one valve stroke in the closed direction, a loud noise was noted by personnel near the valve. On November 16, VY disassembled the valve and found that the stem had broken, approximately two inches above the backseat. The valve was repaired with the assistance of a vendor representative and was returned to service on November 21.

Ultrasonic testing on November 20 revealed a crack indication in the stem of RHR-65A. At the time of this discovery, the plant was in the shutdown condition and no RHR system functions were required to be operable by the TS. The "A" RHR subsystem was declared inoperable and the valve stem was replaced. After post maintenance testing, valve RHR-65A was declared operable on December 1. The cracked stem from RHR-65A and the broken stem from RHR-65B were both sent off-site for metallurgical evaluation.

The bypass valves for the RHR heat exchangers are normally open when their respective RHR subsystems are in the standby alignment for low pressure coolant injection (LPCI). The bypass valves are 20-inch globe valves and the LPCI injection flow enters the valve body beneath the valve disk. With a broken stem, the valve may operate similar to a stop-check valve. However, the valve stem failure for RHR-65B raises two potential safety issues. First, the valve disk could have become lodged in the valve body and affected the LPCI injection flow. Second, the valve disk may not have been able to fully close and therefore could have reduced the ability to remove decay

heat from the containment. A preliminary NRC risk assessment has determined that this second potential failure mode would be the more risk significant scenario.

Pending the results of VY's root cause investigation and metallurgical evaluation, the RHR-65B valve stem failure and RHR-65A valve stem crack will be tracked as an inspector follow-up item. **(IFI 05000271/1999009-02: Root Cause For RHR Valve Stem Failure)**

c. Conclusions

VY's initial response and evaluation of a Residual Heat Removal valve performance anomaly was not thorough and did not have adequate engineering involvement. While following up on NRC questions, engineering personnel identified that the initial evaluation was incorrect and that the valve was degraded. When valve RHR-65B was disassembled, VY discovered its stem was broken.

A valve stem failure for one RHR heat exchanger bypass valve (RHR-65B) and a crack in the redundant train's valve stem (RHR-65A) were discovered during the 1999 refueling outage. Pending the results of the licensee's root cause investigation and metallurgical evaluation, this issue will be tracked as an inspector follow-up item.

The design capability of the RHR heat exchanger bypass valves (based on torque switch settings) was not appropriately incorporated into two operating procedures as required by Quality Assurance requirements for Design Control (10CFR50, Appendix B, Criterion III). This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy, (NUREG 1600, November 9, 1999). This violation was entered in VY's corrective action program as ER 99-1493 and ER 99-1427.

M2.2 Inservice Inspection (ISI)

a. Inspection Scope (73753)

This inspection was performed to confirm the plans, schedules, and activities for the current ISI interval (second period, third interval) were in compliance with the requirements of American Society of Mechanical Engineers (ASME) Section XI, 1986 edition (no addenda) and 10CFR50.55a(g). Areas inspected included ASME Section XI ISI program coverage, self assessment of ISI program activities, weld examination schedules, qualifications and certifications of nondestructive examination (NDE) personnel, NDE procedures, results of NDE, and oversight of NDE contractor activities. The inspector observed performance of NDE including manual ultrasonic (UT) examination and selected portions of remote in-vessel visual inspection (IVVI) and automated UT examinations.

b. Observations and Findings

VY has NDE contractors perform ISI and IVVI using VY test procedures, certain test equipment, and expendable products (couplant, cleaners, penetrant and developers).

VY provides oversight that includes review and approval of personnel qualifications, surveillance of field testing, and review and acceptance of test results. The inspector reviewed portions of the ultrasonic, penetrant and magnetic particle test procedures in use by NDE personnel and found them to be in accordance with the intent of the ASME Code requirements, and adequate for the NDE tasks performed. NDE contractor personnel were site-specific trained and qualified in the use of the VY test procedures. NDE test reports examined by the inspector had been reviewed and accepted by the VY ISI coordinator and reviewed by the authorized nuclear inservice inspector (ANII).

The inspector observed the UT system calibration and subsequent examination of the weld of the N1B recirculation (suction) nozzle to the reactor pressure vessel. The inspector identified that the calibration block used to perform the calibration of the UT system was not accurately represented by controlled drawing, RV 3. The location of the drilled hole calibration reflectors shown on the drawing were actually at the opposite end of the calibration block. VY initiated ER 99-1688, "Calibration Block RV-3, Drawing Configuration," identifying the discrepancy and entering the item in the corrective action program. The inspector determined that the location of the drilled hole reflectors did not have an adverse effect on the outcome of the calibration process or the actual test process and inspection results.

The inspector reviewed the certification records of the examiners performing this NDE and found they were appropriately qualified based on having fulfilled the education, training, examination, and experience requirements specified in SNT-TC-1A, "Recommended Practice for Non-destructive Testing Personnel Qualification and Certification." Additionally, the inspector selected completed inservice inspection reports for visual (VT), liquid penetrant (PT), magnetic particle (MT) and UT processes and determined that examination results, evaluation, and corrective action (if any) were properly recorded. The documentation of the examination process and evaluation of indications was satisfactory.

The inspector reviewed portions of the in-vessel visual inspection of the circumferential welds of the jet pump assemblies, core spray piping and tee box welds, core plate rim hold down bolts, shroud tie rods, and associated hardware. In addition, the inspector observed the remote automatic ultrasonic testing of the circumferential welds in the mixer, diffuser and adapter sections of the jet pumps. No deficiencies were noted in this NDE activity.

VY has a program which governs the development and implementation of departmental self-assessment activities in all functional areas. The program is outlined in VYP 115, "Vermont Yankee Self-Assessment Policy," and in AP 6005, "Functional Area Assessment Development." The program provides for the objective evaluation of essential characteristics as a measure of performance. The inspector evaluated oversight of contractor NDE activities by reviewing the results of self-assessments performed in the Project Engineering functional area which includes ISI activities. No nonconforming conditions were noted as a result of the assessments. In addition, VY quality assurance (QA) performs periodic surveillance of field NDE activities and verifies individual examiner qualifications. The inspector reviewed the surveillance activities for the period October 27, 1999 through November 10, 1999, which involved the observation of seven NDE-related activities. The surveillance reports defined scope,

outlined the activity, and identified test characteristics observed. No significant findings were identified during the surveillance period.

c. Conclusions

Inservice inspection activities during the 1999 refueling outage were performed acceptably and included acceptable ASME program coverage, qualified personnel, approved procedures, proper implementation, appropriate examination documentation, and VY oversight. NDE results were well documented and indications were appropriately recorded and resolved. VY's inspections were thorough and of sufficient extent to determine the integrity of the components inspected.

M7 Quality Assurance in Maintenance Activities

M7.1 Oversight of Contractor Activities During the 1999 Refueling Outage

a. Inspection Scope (37550)

The inspector reviewed the VY oversight and control of contractor activities during the current refueling outage. This inspection was in response to problems that occurred during the 1998 refueling outage including: weak radiological practices, inattention to procedure details, and poor industrial safety habits. The review included the examination of programmatic documents, tours of the plant, and interviews with station personnel.

b. Observations and Findings

The individual issues of poor performance were corrected during the 1998 outage, but long term corrective actions to prevent recurrence were not developed until prior to the 1999 refueling outage. The corrective actions included: 1) the development or revision of program and administrative procedures for the control of contractors; 2) the assignment of a VY designated representative/supervisor to each major contract group/activity; and 3) an emphasis by the quality assurance department on contractor activities.

The inspector reviewed the program for overall contractor control, the administrative procedures for procurement of contractors, and oversight of contractors both on-site and off-site. The concept to assign specific VY personnel to oversee activities appears to have been a good initiative. The inspector reviewed a summary of the QA observations and the VY Event Report database since the beginning of the outage; there were few issues directly or indirectly related to contractor activities.

The inspector concluded that the new program for control of contractor activities improved the performance and attention to detail by most of the station personnel. The new program was effective in identify and correcting errors before the work was completed. For instance, during some valve machining in the hot-shop, the contractor inadvertently failed to obtain a nondestructive examination (NDE) of a weld prior to continuing with the work package. The VY designated representative identified the

missed NDE, stopped the work, discussed the technical aspects of the missed NDE with the appropriate personnel, and initiated an ER to document the issue. The inspector considered this to be a good example of identifying and correcting problems under the improved oversight of contractor activities.

c. Conclusions

Overall, the control of contractors and station personnel regarding procedure adherence has improved since the last refueling outage. The additional oversight helped to ensure the proper identification and correction of problems. A review of QA observations and Event Reports from the outage found few issues directly or indirectly related to contractor activities. Good oversight was provided by the VY designated representatives and frequent QA observations.

M8 Miscellaneous Maintenance Issues

M8.1 In-office Review of LERs Related to Maintenance (90712)

An in-office review of a Licensee Event Report (LER) was performed to assess whether further NRC action is required. The adequacy of the overall event description, immediate actions taken, cause determination, and corrective actions were considered during this review.

(Closed) LER 05000271/1999004-00: Inadequate Communication Results in a Work Team Rendering HPCI System Inoperable During Minor Modification Installation

On October 5, 1999, with the reactor operating at 99 percent power, the high pressure coolant injection (HPCI) system was rendered inoperable due to a leak in the lube oil system. During installation of a minor modification, a plant worker leaned on a HPCI oil line, causing the tubing to bend and creating a small oil leak. The problem was immediately reported to the control room, the HPCI system was declared inoperable, and a seven day limiting condition for operation was entered. The lube oil system tubing was repaired and the system was returned to an operable status within 12 hours. No violation of NRC requirements occurred and the unexpected loss of a single train safety system was reported in accordance with 10CFR50.72. This LER is closed.

III. Engineering

E8 Miscellaneous Engineering Issues

E8.1 (Closed) IFI 05000271/1998006-02: EPG/SAG Transition Criteria for Containment Floodup

a. Inspection Scope (92903)

The inspector reviewed an issue concerning the transition criteria in the Vermont Yankee (VY) Emergency Operating Procedures (EOPs) related to containment flooding following a design basis loss of coolant accident (DB-LOCA). The inspection included

interviews with the EOP Coordinator, and reviews of the associated EOPs, the Boiling Water Reactor Owners' Group (BWROG) guidance documents, the safety evaluations, and Event Reports. During the review, the inspector also resolved concerns with portions of the hardened vent modification installed in 1992.

b. Observations and Findings

During an NRC special inspection conducted in 1998, the inspector identified that if Vermont Yankee experienced a DB-LOCA, the VY EOPs would have required flooding of the containment based on the inability to maintain reactor water level greater than the top-of-active-fuel (TAF). No consideration was given as to what emergency core cooling systems (ECCSs) were available for injection.

The VY EOPs were developed using Revision 4 of the BWROG emergency procedure guidelines (EPGs). The VY Final Safety Analysis Report (FSAR), Chapter 14.6.3, describes a DB-LOCA as a complete circumferential break of one of the recirculation loop pipelines inside of containment. A safety design of the BWR reactor vessel is that, following a DB-LOCA with minimal ECCS equipment available, the reactor water level would be maintained at the height of the suction side of the jet pumps (⅔ core height). This inherent feature allows for sufficient cooling of the fuel, but it implies that the water level would never be at the TAF during a DB-LOCA.

The NRC Safety Evaluation for Revision 4 of the EPGs stated that "... each BWR licensee who wished to use Revision 4 of the EPG should assure that the EPGs will not impact its licensing basis." The containment flooding procedure directed operators to introduce external sources of water into the primary containment, and vent the reactor pressure vessel (RPV) to allow water to enter the RPV. The unfiltered venting of the RPV would have been performed irrespective of offsite dose rates. As such, containment flooding constituted a change to the licensing basis and should have been evaluated in accordance with 10CFR50.59 to ensure that the change did not involve an unreviewed safety question (USQ).

The inspector discussed whether containment flooding at the TAF constituted a USQ, which required NRC review and approval, with individuals from the NRC's Office of Nuclear Reactor Regulation (NRR). The conclusion was that the changes to the EOPs to flood containment, although consistent with the Owners' Group recommendations, did constitute a USQ which required NRC approval. Specifically, when evaluated using the source term inside containment as assumed in the licensing basis analysis for a DB-LOCA, the dose consequences with containment flooding would be increased. Further, containment flooding created the possibility for new equipment malfunctions such as the failure of the standby gas treatment system ductwork. The failure to perform a safety evaluation for a change to the licensing basis and the failure to obtain NRC approval prior to implementation of a change involving a USQ constitutes a violation of 10CFR50.59.

The situation was created in 1990, when Revision 4 of the BWROG EPGs was adopted into the VY EOPs. Since this issue was first identified, the BWROG revised the EPGs to include severe accident guidelines (SAGs). The new EPGs/SAGs defined new criteria for initiation of containment flooding, including a definition for a minimum steam

cooling reactor pressure vessel water level (MSCRWL). The MSCRWL is between the TAF and the ⅔ core height. If level is greater than the MSCRWL, and at least one injection subsystem is available and running, containment flooding is not required. The inspector confirmed that the revised VY EOPs were consistent with the BWROG's EPG/SAG, and the VY licensing and design basis.

VY reported that a realistic analysis revealed that the dose consequences would justify containment flooding. In addition, the undesirable consequences associated with the containment flooding occur only after a DB-LOCA, which is a low probability event; as such, the overall risk is very low. Due to the overall low risk significance of containment flooding, this violation of 10CFR50.59 is categorized at Severity Level IV and is being treated as a Non-Cited Violation (NCV), consistent with Section VII.B.1.a of the NRC Enforcement Policy, (NUREG 1600, November 9, 1999). This violation was entered in VY's corrective action program as ERs 97-0273, 97-0479, and 97-1328.
(NCV 05000271/1999009-03: Failure to Perform Safety Evaluation During 1990 EOP Changes)

In addition to the above, VY identified a related issue, in that the hardened vent rupture disc would have ruptured during the containment flooding event. The rupture disc, a passive relief device, was installed in April 1992 in response to NRC Generic Letter 89-16. VY ER 97-0306 described the fact that the effects of the EOP containment flooding scenario were not considered during the design and installation of the hardened vent modification. The immediate correction actions included closing of the downstream isolation valve and performing the required operability determination. Subsequently, an FSAR design change and safety evaluation were processed to change the valve from normally open to normally closed.

The inspector reviewed the changes and found them adequate. Nonetheless, the failure to consider the effects of containment flooding, directed by the EOPs, in review of the 1992 hardened vent modification is a violation of 10CFR50, Appendix B, Criterion III, "Design Control." This Severity Level IV violation is being treated as an NCV, consistent with Section VII.B.1.a of the NRC Enforcement Policy, (NUREG 1600, November 9, 1999). This violation was entered in VY's corrective action program as ER 97-0306.
(NCV 05000271/1999009-04: Inadequate Design Control for 1992 Hardened Vent Modification)

c. Conclusions

An NRC review of Emergency Operating Procedure actions for containment flooding identified that VY's 1990 incorporation of generic guidance resulted in an unreviewed safety question (USQ). VY's implementation of this change, without prior NRC approval, is a violation of 10CFR50.59. The procedures have since been appropriately revised and this error is not considered indicative of current licensee performance. Due to the overall low risk significance of containment flooding, this violation is being treated as a Severity Level IV, Non-Cited Violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy, (NUREG 1600, November 9, 1999). This violation was entered in VY's corrective action program as ERs 97-0273, 97-0479, and 97-1328.

VY's failure to evaluate the effect of containment flooding during a 1992 design change for a containment hardened vent is a violation of Quality Assurance requirements (10CFR50 Appendix B, Criterion III, Design Control). Potential problems were resolved by subsequent procedure changes and this error is not indicative of current engineering performance. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy, (NUREG 1600, November 9, 1999). This violation was entered in VY's corrective action program as ER 97-306.

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Radiological Controls (Program Changes)

a. Inspection Scope (83750)

The inspector reviewed selected radiological controls program changes since the previous inspection in this area. Areas reviewed included organization and staffing, facilities and equipment, and procedure changes. The inspector reviewed work in progress, reviewed applicable documentation, and interviewed cognizant personnel.

b. Observations and Findings

There were no program changes identified that adversely affected the radiation protection program. VY was effectively implementing its new electronic dosimetry system during the outage to provide realtime monitoring of radiation exposure. VY was monitoring ongoing radiological work activities to identify anomalous exposure accumulation and had established additional in-field radiological controls work stations to control and monitor work. VY established a defined radiological controls organization structure for the outage and was using appropriately trained and qualified personnel to provide oversight of radiologically significant work activities.

VY implemented significant improvement in its planing and preparation of outage work through implementation of a new outage work planning and control procedure. Overall work planning and coordination was observed to be considerably improved. One exception was the pre-job briefing for initial torus entry on November 2, 1999. Among other concerns, all appropriate individuals were not in attendance at the meeting, there was confusion as to which torus ingress point was to be used, and there was confusion as to how work activities were to be conducted. This task was suspended pending re-planning and coordination. An event report was initiated to document the matter. Also, the licensee initiated reviews of planned outage activities to identify other potentially similar issues.

Of particular note was VY's monitoring of emergent work via a scope control committee. Radiological controls management personnel attended scope control meetings and initiated planning and preparation for emergent work to be performed. No changes were identified that adversely affected radiation protection program performance.

c. Conclusions

VY continued to maintain a good radiation protection program. In general, improved work planning and controls were well implemented. Emergent work was formally evaluated and approved for addition to outage scope. No changes were identified that adversely affected radiation protection program performance.

R1.2 Refueling Outage Radiological Controls (ALARA) Planning and Performance

a. Inspection Scope (83750)

The inspector selectively reviewed ALARA planning and preparation efforts and implementation for the refueling outage. The inspector reviewed radiological control records; interviewed licensee representatives relative to outage planning; and observed activities to determine the effectiveness of planning, preparation, and management oversight for radiologically challenging work activities. The inspector reviewed (either through review of documentation or direct observation) selected work activities that had the potential for creating radiological hazards (e.g., refueling activities, reactor water cleanup pump work, main steam isolation valve work, control rod drive removal, reactor core isolation cooling (RCIC) system work, turbine work, and reactor recirculation pump seal replacement).

b. Observations and Findings

VY performed overall effective planning and preparation for outage radiological work activities and implemented good efforts to reduce personnel occupational exposure for work activities to as low as is reasonably achievable. VY used mock-up training and remote cameras in an effort to reduce aggregate exposure values. VY modified its drywell shielding installation to provide for increased shielding. Dose rates in selected areas were observed to be significantly lower than the previous outage. Work activities for tasks that could result in elevated aggregate personnel radiation exposure were controlled by radiation work permits with detailed ALARA reviews and clearly defined radiation exposure reduction methods.

VY performed a review of ongoing work to identify anomalies and implement enhanced oversight or controls to minimize occupational exposure. VY ALARA Engineers identified elevated radiation levels on the morning of November 3, 1999, in bay 12 of the 213-foot elevation of the torus in an area where work was being conducted on a RCIC line. However, the area was not posted to inform workers of the elevated radiation dose rates, workers were not informed of the elevated levels, and radiation protection personnel, covering work activities in the area, were also not informed of the matter. Radiation surveys used to brief workers in the area did not depict the elevated dose rates. Subsequently workers entered and worked in the area. The workers were aware of elevated radiation levels on the torus and were instructed to stand away from the torus. However, workers were under the perception that dose rates along the torus were on the order of 8 mR/hr and periodically stood in the area to observe work in the overhead area. Subsequent inspector surveys in the area identified workers standing in dose rates of 50 mR/hr. Also, contact dose rates in material storage areas accessed by workers were found to measure about 140 mR/hr. The workers were unaware of these

dose rates. A recent radiation survey for the area was incorrectly identified as the survey of torus bay 16. The inspector informed radiation protection personnel who immediately resurveyed the area, posted precautionary signs to alert workers standing in the area of the elevated radiation dose rates, and briefed the workers on radiation dose rates. VY placed this matter into its corrective action system and resurveyed the accessible portions of the torus outer shell to identify similar concerns.

10CFR19.12 requires that workers be informed of precautions and procedures to minimize exposure and be kept informed of the storage, transfer, or use of radiation or radioactive material. The failure to adequately inform workers of the magnitude of radiation dose rates in their work areas for purposes of exposure minimization is a violation of 10CFR19.12. This Severity Level IV violation is being treated as an NCV, consistent with Section VII.B.1.a of the NRC Enforcement Policy, (NUREG 1600, November 9, 1999). This violation was entered in VY's corrective action program as ER 97-1503. **(NCV 05000271/1999009-05: Failure to Adequately Inform Workers of Radiation Dose Rate)**

c. Conclusions

Overall, VY implemented an effective ALARA program. There was effective planning and preparation for outage radiological work activities. VY implemented good efforts to reduce personnel occupational exposure for work activities to as low as is reasonably achievable.

One violation for the failure to inform workers of elevated radiation dose rates in their work area was identified by the NRC. This Severity Level IV violation is being treated as an NCV, consistent with Section VII.B.1.a of the NRC Enforcement Policy, (NUREG 1600, November 9, 1999). The violation was entered in VY's corrective action program as ER 97-1503.

R1.3 Refueling Outage Radiological Controls (Internal and External Exposure Controls)

a. Inspection Scope (83750)

The inspector reviewed records, interviewed cognizant licensee personnel, and observed occupational exposure control practices during licensee work activities and tours of the RCA. The inspector reviewed high radiation area controls, general radiological posting, implementation of the radiation work permit program, and implementation of the dosimetry program. The inspector toured the drywell, refueling floor, and reactor building and observed ongoing activities and radiological conditions. The inspector selectively made independent radiation measurements to verify licensee results and reviewed selected work activities that had the potential for creating radiological hazards (e.g., reactor head removal, refueling activities, reactor water cleanup pump work, recirculation pump seal replacement, in-vessel inspection, main steam isolation valve work, turbine and heater bay work, and control rod drive removal).

b. Observations and Findings

Overall, VY made appropriate radiological surveys to support planning and preparation of work and made appropriate job-coverage surveys to monitor ongoing work. VY provided and used calibrated and checked survey instrumentation for radiological surveys. Personnel dosimetry was properly issued, worn, and moved to points of highest expected radiation exposure of the body. Multiple dosimetry was used as appropriate. Radiation work permit ALARA reviews provided overall good guidance for workers and were properly implemented. Selective verification identified that workers were properly signed in on their assigned RWPs. No significant unplanned personnel external or internal exposures were identified. VY implemented its radiological controls program for visitors and declared pregnant females.

Engineering controls were effectively used to minimize airborne radioactivity. Selected workers, who wore respiratory protective equipment, were verified to have received training, fit testing, and medical certification to wear the equipment. VY evaluated low level intakes of radioactive materials using data from air samples and whole body counters, as appropriate. No significant airborne radioactivity was identified and no individuals sustained any significant airborne radioactivity intake. VY used an effective derived airborne radioactivity concentration (DAC) which took into consideration hard to detect radionuclides.

Access points to areas of elevated radiation levels or areas exhibiting contamination were properly posted and barricaded. Appropriate access controls were implemented for High Radiation Areas, including those areas meeting criteria to be locked. VY implemented its locked High Radiation Area access control program.

c. Conclusions

Applied radiological controls for ongoing work activities were generally well implemented. No significant unplanned personnel external or internal exposures were identified. No significant airborne radioactivity was identified and no individuals sustained any significant airborne radioactivity intake.

R1.4 Refueling Outage Radiological Controls (Control of Radioactive Materials and Contamination)

a. Inspection Scope (83750)

The inspector selectively reviewed radioactive material and contamination control practices, including the adequacy of supply, maintenance, calibration, and performance checks of survey and monitoring instruments; the use of personal contamination monitors and friskers; and application of hot particle contamination monitoring.

The evaluation of licensee performance in this area was based on observations during station tours, discussion with cognizant personnel, and review of documentation.

b. Observations and Findings

Radioactive material was properly labeled, stored, and controlled. Contamination monitoring equipment was observed to be operable, within calibration, and properly

used by personnel. Overall, radiation and contamination surveys were observed to be comprehensive and detailed, but some documentation weaknesses were identified. VY used appropriate contamination control techniques and used whole body monitors to monitor personnel for contamination when exiting the radiological controlled area. Personnel were also whole body monitored for contamination with a different type of contamination monitor when exiting the Security Building. VY was evaluating its in-plant source term to ensure the personnel monitors remained adequate for changes in radionuclide mix. Personnel identified as contaminated were decontaminated, as appropriate, with appropriate dose evaluations conducted. There were minimal instances of personnel contamination during the outage and dose assessments were conducted as appropriate. There were no significant doses associated with personnel contamination.

c. Conclusions

Overall, VY implemented a good radioactive material and contamination control program. Radioactive material was properly labeled, stored, and controlled; contamination monitoring equipment was operable, within calibration, and properly used by personnel. Radiation and contamination surveys were observed to be generally comprehensive and detailed with some exceptions noted. There were minimal instances of personnel contamination during the outage and no significant dose consequences.

R7 Quality Assurance in RP&C Activities

R7.1 Audits, Surveillances, and Self-Assessment Activities

a. Inspection Scope (83750)

The inspector selectively reviewed quality assurance activities including, as appropriate, audits, surveillances, and self-assessment activities. The inspector also attended event report screening meetings.

b. Observations and Findings

VY implemented an active audit and surveillance program. Quality Assurance (QA) personnel were observed in the field auditing ongoing radiological activities and initiating event reports for identified findings. QA personnel were developing a data base to allow ease in identification of areas for performance enhancement.

In addition, supervisors were conducting tours to review ongoing activities. Performance data from the supervisor tours was incorporated into a data base for review and evaluation by management.

Event reports were initiated at low threshold levels and appropriately screened by management at ER screening meetings.

c. Conclusions

VY implemented overall good oversight of ongoing radiological controls activities. There was good in-field presence by QA personnel and supervisors, who conducted daily observations of ongoing work activities. Performance assessment findings were provided to plant management.

R8 Miscellaneous RP&C Issues

R8.1 Plant Tour Observations

a. Inspection Scope (83750, 71707)

The inspector made various tours of the radiological controlled areas including the turbine building, turbine heater bays, drywell, reactor building, refueling floor, and radwaste areas. The inspector reviewed general housekeeping and station material conditions.

b. Observations and Findings

Overall housekeeping was very good. Walkways were unobstructed and potentially contaminated materials were clearly marked and segregated. Housekeeping conditions in the drywell were generally good. Contaminated areas (e.g., control rod drive travel paths) were promptly decontaminated after work completion to allow ease of access. No buildup of combustibles was observed.

c. Conclusions

VY implemented overall good housekeeping within radiological controlled areas.

V. Management Meetings

X1 Exit Meeting Summary

The resident inspectors met with licensee representatives periodically throughout the inspection and following the conclusion of the inspection on January 18, 2000. In addition, inspectors from the NRC Region I office met with licensee representatives at the conclusion of their on-site reviews. At these meeting, the purpose and scope of the inspections were reviewed, and the preliminary findings were presented. The licensee acknowledged the preliminary inspection findings.

The inspector asked the licensee whether any material examined during the inspection should be considered proprietary. No proprietary information was identified.

LIST OF ACRONYMS USED

ALARA	As Low as Is Reasonably Achievable
ANII	Authorized Nuclear Inservice Inspector
AP	Administrative Procedure
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CFR	Code of Federal Regulation
CRDM	Control Rod Drive Mechanism
DAC	Derived Airborne Radioactivity Concentration
DB-LOCA	Design Basis Loss of Coolant Accident
DR/S	Designated Representative / Supervisor
ECCS	Emergency Core Cooling System
EDCR	Engineering Design Change Request
EOP	Emergency Operating Procedure
EPG	Emergency Procedure Guideline
ER	Event Report
FME	Foreign Material Exclusion
FSAR	Final Safety Analysis Report
GL	Generic Letter
GPM	Gallons per Minute
HCU	Hydraulic Control Unit
HPCI	High Pressure Coolant Injection
I&C	Instrument and Controls
IFI	Inspector Followup Item
IR	Inspection Report
ISI	Inservice Inspection
IVVI	In Vessel Remote Visual Examination
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LPCI	Low Pressure Coolant Injection
LPRM	Local Power Range Monitor
mR/hr	Millirem per Hour
MSCRWL	Minimum Steam Cooling Reactor Pressure Vessel Water Level
MT	Magnetic Particle
NDE	Nondestructive Examination
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
NRR	NRC Office of Nuclear Reactor Regulation
OP	Operating Procedure
PSIG	Pounds per Square Inch - Gauge
PT	Liquid Penetrant
QA	Quality Assurance
RCIC	Reactor Core Isolation Cooling
RFO	Refueling Outage
RHR	Residual Heat Removal

LIST OF ACRONYMS USED

RPM	Radiation Protection Manager
RPV	Reactor Pressure Vessel
RWP	Radiation Work Permit
SAG	Severe Accident Guideline
SFP	Spent Fuel Pool
SLC	Standby Liquid Control
SRM	Source Range Monitor
SRO	Senior Reactor Operator
STP	Special Test Procedure
TAF	Top of Active Fuel
TS	Technical Specification
USQ	Unreviewed Safety Question
UT	Ultrasonic Examination
VT	Visual Inspection
VY	Vermont Yankee
WO	Work Order

Attachment 2

ITEMS OPENED, CLOSED, OR DISCUSSED

OPENED

IFI 05000271/1999009-02:	Root Cause For RHR Valve Stem Failure (page 11)

CLOSED

LER 05000271/1999004-00:	Inadequate Communication Results in a Work Team Rendering HPCI System Inoperable During Minor Modification Installation (page 14)
IFI 05000271/1998006-02:	EPG/SAG Transition Criteria for Containment Floodup (page 15)

NON-CITED VIOLATIONS

NCV 05000271/1999009-01:	Failure to Translate MOV Design into Operating Procedures (page 10)
NCV 05000271/1999009-03:	Failure to Perform Safety Evaluation During 1990 EOP Changes (page 16)
NCV 05000271/1999009-04:	Inadequate Design Control for 1992 Hardened Vent Modification (page 17)
NCV 05000271/1999009-05:	Failure to Adequately Inform Workers of Radiation Dose Rate (page 19)