NUCLEAR REGULATORY COMMISSION
REGION I

Report No.	95-19
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:	July 16 - August 21, 1995
Inspectors:	William A. Cook, Senior Resident Inspector Paul W. Harris, Resident Inspector John T. Shedlosky, Project Engineer

Approved by: Richard J. Conte 9/13/95
Richard J. Conte Date
Chief, Reactor Projects Section 3A

Scope: Station activities inspected by the resident staff this period included Operations, Maintenance, Engineering, Plant Support, and Safety Assessment and Quality Verification. Backshift inspections including weekend activities amounting to 13.5 hours were performed on July 20, 30, 31 and August 7, 8, and 10. Interviews and discussions were conducted with members of Vermont Yankee management and staff as necessary to support this inspection.

Findings: An overall assessment of performance during this period is summarized in the Executive Summary. Two non-cited violations were identified. One involved the inadequate implementation of a fire watch and another involved inadequate inservice testing of core spray and residual heat removal keep fill check valves. One unresolved item involving performance monitoring was identified.

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EXECUTIVE SUMMARY

Vermont Yankee Inspection Report 95-19

Safety Assessment/Quality Verification

Vermont Yankee continued to operate safely. However, significant programmatic and hardware deficiencies were self-identified in the 10 CFR Part 50 Appendix R and Fire Protection Programs. Based upon hardware and program deficiencies, it was apparent that program and implementation responsibilities were not effectively integrated between VY departments involved. We note that a similar type ownership observation was made with respect to the integration of service water (SW) activities (reference IR 95-06, section 3.1).

Two small pin hole leaks in the SW system were identified and satisfactorily addressed. An engineering evaluation provided appropriate assurance that this section of SW piping would remain structurally sound for the duration of the current operating cycle. The temporary modification (pipe clamp) initiated for this repair effort was installed in a configuration different than that approved by the onsite safety review committee. Although VY determined that the installed configuration was not a system safety concern, deficiencies identified with the installation of the temporary pipe clamp represented poor procedural adherence and a lack of clarity in the installation instructions.

Operations

Plant staff trending and evaluation of a core spray system piping slow pressurization problem and a recirculation pump seal differential pressure anomaly was found to be comprehensive. Although efforts to prevent lightning-induced damage to plant equipment have been taken, VY continues to voluntarily remove TS radiological effluent instrumentation from service to prevent damage. The recurrence of high area room temperatures and service water process temperatures caused by elevated ambient and resultant higher than average Connecticut River temperatures indicated the lack of an integrated approach in dealing with warm weather and was viewed as an operational performance weakness. The inadvertent omission of a surveillance test procedure step caused the "A" emergency diesel generator (EDG) to trip; but, more importantly, the immediate EDG post-trip review of that event lacked structure and depth of the assessment of human performance aspects.

Maintenance and Surveillance

The low priority assigned to a work order to repair a degraded Appendix R fire wrap in the Cable Vault resulted in the progressive deterioration of this fire barrier due to foot traffic and its subsequent inoperable condition. The on-line maintenance of the standby liquid control system and the diesel fire pump were well controlled and appropriately justified.

The implementation of the RBCCW-SW heat exchanger performance monitoring was poor as evidenced by the absence of pre-evolutionary reviews and procedural guidance or instructions.

Vermont Yankee management and NRC staff plan to further review the adequacy of administrative controls for the conduct of system manipulations for the purposes of performance monitoring or data taking (Unresolved 95-19-01).

Engineering

Inspectors observed the timely and proper resolution of inadequate inservice testing of safety system check valves. The corrective actions were comprehensive and interim non-intrusive testing (radiography) was appropriately implemented, pending a permanent fix. This Inservice Testing deficiency was dispositioned as a non-cited violation.

As mentioned above, programmatic and hardware deficiencies in the 10 CFR Part 50, Appendix R, Fire Protection strategy were self-identified and compensatory actions were taken. Significant time and resources were being expended by the VY staff to appropriately identify and resolve all concerns. The inspectors identified that appropriate management attention has been focused on this issue and that the corrective action plan implemented appears to be comprehensive.

Plant Support

The radiation protection staff ALARA review for the Appendix R fire watches was slow, as well as, the issuance of a specific radiation work permit for the fire watches. During the emergency preparedness drill, VY demonstrated generally good communications and coordination of resources, and an effective post-drill critique was conducted. Unlike previous successes in promptly identifying abnormal reactor coolant chemistry conditions, the chemistry staff was not timely in identifying and evaluating the recent increase in copper concentrations.

Inadequate post instructions to a fire watch, established to compensate for Appendix R deficiencies, resulted in the failure of that fire watch to continuously monitor all areas within the reactor building fire zone of concern. This event was dispositioned as a non-cited violation.

TABLE OF CONTENTS

Note: Procedures from NRC Inspection Manual Chapter 2515, "Operating Reactor Inspection Program" are used as inspection guidance and are parenthetically listed.

DETAILS

1.0 SUMMARY OF FACILITY ACTIVITIES

Vermont Yankee (VY) operated at 100 percent rated reactor power for the most of this inspection period. On July 25, reactor power was reduced to 95 percent in accordance with Technical Specifications when VY declared the "B" safety relief valve inoperable during surveillance testing. A second power reduction to 90 percent occurred on August 7, when higher than normal vibrations were observed on the main turbine.

2.0 OPERATIONS (71707)

2.1 Operational Safety Verification

The inspectors verified adequate control room (CR) staffing, adherence to procedures and technical specification (TS) limiting conditions for operations (LCO), operability of safety systems, status of CR annunciators, and functionality of emergency core cooling systems. The inspectors verified that the control room operators (CROs) were cognizant of active LCOs, maintenance in progress, and deficiencies noted the previous shift. The inspectors noted that emergency operating procedure entry conditions for high area temperatures (Section 2.8) were also understood by the staff and that the plant process computer was used to monitor these conditions.

The inspectors observed the conduct of CR pre-shift briefings and the CRO watch turnovers. Plant information such as maintenance, surveillance, and equipment out of service was discussed in appropriate detail. The conduct of watch reliefs included log, surveillance and tagout reviews. Control room panel walkdowns were also conducted to familiarize the oncoming operators with the status of systems and components. The inspector noted that the operators appropriately discussed the status of the service water (SW) pinhole leaks and Appendix R deficiencies and fire watches. Routinely, the inspectors observed that Operations Department management provided oversight during the CR briefings.

The inspector reviewed the equipment safety tagouts for maintenance conducted on the Standby Liquid Control, "B" reactor water cleanup pump, and SW systems and noted no discrepancies. The components were properly isolated for maintenance to assure personnel and equipment safety. The inspector noted that the SW tagout also provided assurance that secondary containment integrity would be maintained during the maintenance.

2.2 Vernon Tie Line Outage

On July 15, at 6:45 p.m., the Vernon Tie Line became inoperable due to electrical problems at the Vernon hydroelectric station. The Vernon Tie Line is a dedicated electrical power supply from the hydroelectric station to VY safety-related 480 Vac electrical distribution. This offsite electrical supply is credited as reliable and independent in the VY station blackout analysis. It is also a major factor in limiting core damage frequency as described in the VY Individual Plant Examination.

The inspector noted appropriate operational actions in response to this occurrence. Management reviewed the event, an Event Report (ER) was initiated, and entry into the applicable TS was made to track the out of service time. During the approximate 3.5 hours the tie line was inoperable, both emergency diesel generators were operable and other offsite power supplies were available. The July 15 occurrence resulted in a negligible increase in Vernon Tie Line unavailability. The loss and restoration of the tie line was out of VY's control because the electrical fault occurred outside VY's electrical distribution system.

2.3 Reactor Water Cleanup System Operation

The inspector examined Operating Procedure (OP) 2112, Reactor Water Cleanup System, and discussed reactor water cleanup (RWCU) system operation with CROs. The inspector was particularly interested in determining if operators experienced significant flow perturbation while shifting between filter demineralizers and if flow dependent isolation signals were temporarily bypassed to facilitate this evolution. The inspector determined that the flow isolation signal is not bypassed and that the demineralizer bypass valve (12 MOV-74) is throttled open to stabilize RWCU system flow during demineralizer shifting.

Based on discussions with the operators, the inspector determined that RWCU filter demineralizer shifting was routinely conducted without incident on an approximate weekly basis. However, in recent weeks RWCU system cleanup filter demineralizer resin backwash and pre-coating evolutions have met with some difficulties, resulting in overflows of the resin batch tank. The mildly contaminated overflow was appropriately contained and promptly cleaned-up. The cause for these tank overflows was still being investigated at the conclusion of the inspection period. A leaking air supply valve used in the backwash cycle is suspected to be the cause. With the exception of adding new resin to the batch tank, nearly all backwash and pre-coat valve manipulations are automatically timed and controlled. Operators and instrumentation and controls technicians continue to investigate this operational problem.

2.4 Fuel Pool Cooling Systems

The inspector examined operation of the spent fuel pool (SFP) cooling systems via review of the Final Safety Analysis Report (FSAR), Section 10.5; OP 2179, Standby Fuel Pool Cooling, and via discussions with the CROs. The inspector also conducted a partial walkdown of accessible portions of this system. The inspector was particularly interested in the means of level control and monitoring of the SFP. Normal makeup capability is provided by either the condensate transfer system or the demineralized water system. Level is lowered via the SFP filter demineralizer rejection to the condensate storage tank. Remote level indication is provided by level transmitters located in the northwest and southeast corners of the SFP. Spent fuel pool level is maintained manually between the pool high and low levels alarms. Normal level is maintained at approximately 37 feet, two inches. This normal level is about 12 inches below the SFP/reactor building ventilation exhaust ducting which draws approximately 13,200 SCFM from the SFP area. The high level alarm is set at a level approximately six inches below the pool ventilation exhaust ducting. Based upon discussion with the station staff, no recent events have occurred where SFP level has approached the inlet of the ventilation exhaust duct work. Auxiliary operators (AOs) check SFP level and conditions during their once per shift (every 8 hours) rounds. The inspector concluded that appropriate means were available to monitor SFP conditions.

2.5 Safety System Walkdowns

The inspector conducted partial system walkdowns of the standby liquid control, residual heat removal, and core spray systems and verified proper system configuration. Control room panel (CRP) indications accurately represented the positions of valves and breakers in the field. The applicable system operating procedures were also reviewed confirming that the system lineups were appropriate. The material condition of the system components was also examined and found to be very good. For example, components and systems were free of surface corrosion and system leakage, and were readily accessible. In addition, electrical connections were sound and housekeeping in the vicinity of the systems was very good.

2.6 Review of Lightning Strike Preparations

During the summer months, thunderstorms traversed the Connecticut River valley area and, in accordance with OP-3127, "Natural Phenomena," CROs took action to minimize the potential for damage to the main stack radiation monitors, a more susceptible component due to their location in the main stack. Per procedure, operators de-energized the No. 2 low range stack gas monitor (one of two low range gaseous effluent monitors) and the high range stack noble gas effluent monitor No. 3 while the storms were in the vicinity.

Inspector review of the SS's log and OP-3127 and discussions with the CR staff identified that stack gas monitors No. 2 and 3 were removed from service between 8:55 p.m. July 28 and 1:13 a.m. July 29. Appropriate log entries were made and the inspector verified that the TS LCOs were satisfied. With the assistance of an operator, the inspector walked through the process by which the monitors are procedurally removed from service. Closer examination of the operating procedure by the inspector identified that OP-3127 does not refer to the applicable sections of TS (Table 3.2.6 and Table 3.9.2) in the reference section, or caution operators that de-energization of the stack gas monitors may result in an LCO entry. This was brought to the attention of the SS who promptly took action to revise OP-3127. The inspector noted a subsequent Night Order entry highlighting the procedure change to the operations staff.

The inspector concluded that the VY staff took appropriate actions to preclude equipment damage during thunderstorms. The inspector notes that this voluntary removal of Technical Specification equipment from service to prevent damage is done in addition to earlier plant modifications implemented to minimize the electrical transients imposed by lightning strikes. The VY staff continues to evaluate alternatives to improve their plant systems lightning strike protection.

2.7 Monitoring of Reactor Plant Parameters

2.7.1 Core Spray System Pressure

The VY staff continued to trend CS system isolation valve integrity (reference NRC Inspection Report 95-06) and noted during this inspection period that the "B" CS piping pressure was slowly increasing. By the end of the inspection period, both the "A" and "B" CS systems were experiencing a slow pressurization rate of approximately 3.3 to 6.6 psig/day. Operators periodically vent the systems to minimize the pressure build-up.

The inspector reviewed the trend, inspected the accessible portions of the "B" CS system, and reviewed previous corrective actions to improve the leak tightness of CS isolation valves. The CS containment isolation valves and containment check valves were inspected and refurbished during the previous outage. The four downstream CS containment isolation valve disks were drilled to preclude pressure locking (reference Inspection Report 95-03). VY continues to trend the pressure buildup in the "A" and "B" CS subsystems which is occurring at approximately the same rate. The Engineering staff was tasked to assess this condition.

2.7.2 Reactor Recirculation Pump Seals

This period, the pressure drop across the outer seal for the "A" reactor recirculation pump (RRP) remained fairly constant at approximately 535 psig. As documented in Inspection Report 95-17, VY has been monitoring the performance of this seal since CROs noted that pressure was increasing following seal refurbishment conducted during the 1995 refueling/maintenance outage. Seal performance was also evaluated following an unplanned RRP speed excursion (due to a failed capacitor in the speed circuitry) that occurred on July 15. During this speed increase, seal pressure remained relatively constant. The inspector continues to monitor licensee actions involving the "A" RRP seal.

2.8 Warm Weather Preparation and Evaluation

Similar to the summer of 1994, warm weather conditions again caused elevated temperatures within the reactor building (RB) and in the process fluids of several plant systems, resulting in a number of occasions where high ambient air temperatures exceeded emergency operating procedures (EOP) entry conditions. The secondary containment control EOP was entered for elevated temperatures (limit of 106 degrees F was exceeded by 1 degree F) on the 252 foot level and the 280 foot level within the RB. Area ventilation was maximized and the conditions were stabilized. Periodic monitoring was initiated to evaluate changing conditions and corrective maintenance was performed to improve the capacity of RB forced-air coolers. Inspector follow-up of the secondary containment EOP entries determined that the RB temperatures are measured using permanently installed thermocouples placed high overhead in specific locations within the RB, thereby representing a conservative area temperature. Also, ambient air temperatures in excess of

EOP limits does not necessarily mean that equipment degradation will occur. However, prolonged exposure to high temperatures may reduce equipment life expectancy.

Similar to the ambient air temperatures within the RB, the high temperature of Connecticut River water caused higher than normal operating temperatures of some plant process fluids. Elevated ambient air temperature contributed to torus water temperature exceeding its alarm setpoint (~85 degrees F) on several occasions. The higher than normal residual heat removal-service water (RHR-SW) temperature reduced the effectiveness of RHR-SW torus cooling mode of operation necessitating more frequent RHR pump operations. In order to reduce the number of pump start/stop cycles, CROs delayed torus cooling initiation but they frequently monitored torus temperatures by monitoring computer alarm printouts as compensation for a lighted torus trouble annunciator which receives multiple parameter inputs. No design limits were exceeded. However, CROs did not notice (until the alarm condition annunciated) that alternate cooling tower (ACT) deep basin temperature had reached its administrative limit (100 degrees) due to warm river water temperatures. Following this alarm condition, deep basin cooling was initiated. Inspector follow-up of the torus and ACT deep basin elevated temperature events determined that the alarm setpoints were set to provide appropriate margin to the design heat capacity of these safety systems.

The inspector reviewed VY procedures governing the operations of the above systems. In all cases, no reference was made in plant procedures or operating logs as to the FSAR design temperatures of the above systems. The absence of design temperature limits was also noted in the circulating water (CW) procedure, which utilizes the cooling tower system to help maintain acceptable CW discharge temperatures. The inspector determined that the logging of the deep basin and SW supply temperatures is not performed. Trend information, however, was available through the plant process computer.

VY plant management was active in their monitoring and evaluation of the effects of the higher than normal temperatures on the operation of systems and components. Responsibilities were assigned for the evaluation of higher than normal temperature effects on plant operations and the Maintenance Department initiated a review of the forced-air cooler preventive maintenance to assure optimum performance. In addition, the inspector noted that the VY staff concluded elevated CW temperatures may have contributed to higher than normal reactor feedwater copper concentration which resulted the vendor-recommended fuel warrantee limits being exceeded (Section 5.3).

Based on the above observations, the inspector concluded that each individual temperature related problem was of relatively low safety significance due to the temperature limit being exceeded for only a short period of time. Notwithstanding, the number and frequency of elevated temperature-related problems indicated that this aspect of plant operations was not fully integrated to preclude problems. The inspector noted that EQ evaluations were proactive and conducted prior to any adverse conditions occurring and that plant management routinely assessed warm weather problems when they occurred. However, this was the second consecutive summer of above average warm weather which caused elevated temperature conditions within the RB and plant fluid systems and which prompted CROs to enter EOPs for remedial action. The absence of an integrated approach to preclude the recurrence of weather-related problems was a weakness. These observations and overall assessment were acknowledged by station management and a Plant Manager action item was initiated to further evaluate this area.

2.9 "A" Emergency Diesel Generator Surveillance Observations

On July 25, the inspector observed CROs respond to an inadvertent trip of the "A" emergency diesel generator (EDG). The EDG tripped on lockout when the AO at the local control panel did not perform step C.1.16 of OP-4126, Diesel Generator Surveillance. This step required the manual reduction of EDG field flash voltage following field flash. The CROs and the Operations Manager (OM) determined that the cause of the trip was because OP-4126 was not properly implemented. The adequacy of this surveillance procedure was reviewed and an ER was initiated prior to re-test of the "A" EDG. Both EDGs subsequently completed their monthly performance tests satisfactory.

The inspector observed the post-event evaluation in the CR and verified that VY's review was adequate to assure the safe and proper continued surveillance testing of the EDGs. The post-event debrief was attended by all parties involved. During this debrief, the AO stated that he missed performing step 16. At the debrief, VY identified no other problems. The inspector noted that VY classifies OP-4126 as "Reference Use" procedure. This implies that in-hand use was not required, however, the procedure shall be close at-hand and referred to before, during, and after the activity to ensure that correct steps are performed.

The inspector concluded that the post-event CR evaluation provided adequate confidence that the EDG trip was understood. However, the inspector observed a sense of urgency on the part of the CROs to restore the A EDG to its standby (operable) configuration. In this case, had the AO not come forward with his error of omission, the direct cause and evidence of the EDG trip could have been lost due to the CROs actions to quickly restore the EDG to the standby line-up. In addition, the inspector noted a lack of structure and depth to the conduct of the post-event evaluation in the CR. This initial evaluation did not attempt to determined why the AO did not perform the surveillance as written and thereby ensure appropriate initial corrective action was taken. The inspectors discussed the apparent lack of a human factors performance review with station management, who acknowledged the observations and indicated they would incorporate an enhanced (structure and depth) assessment of the VY staff's response to this event via the Event Report process.

3.0 MAINTENANCE

3.1 Maintenance (62703)

The inspectors observed selected maintenance on safety-related equipment to determine whether these activities were effectively conducted in accordance with VY TS, and administrative controls (Procedure AP-0021 and AP-4000) using approved procedures, safe tagout practices and appropriate industry codes and standards. Interviews were conducted with the cognizant engineers and maintenance personnel and vendor equipment manuals were reviewed.

3.1.1 Conduit Fire Wrap Degraded

On July 14, the fire protection staff declared the protective fire wrap on three 10 CFR Part 50, Appendix R, safe shutdown related conduits in the cable vault inoperable. This action required a continuous fire watch be established in the vicinity of the identified conduits as a compensatory measure until the fire wraps could be restored to their tested design configuration. The inspector examined the degraded fire wraps and discussed the post instructions with the continuous fire watch posted in the Cable Vault.

The inspector determined that the VY staff attributed the cause for the fire wrap degradation to have been foot traffic. The inspector also noted that the deterioration of the fire wraps was initially identified during the 1993 operating cycle surveillance and that Work Order 94-9567 was generated, at that time, to repair the barrier.

In summary, the prioritization of Work Order 94-9567 was not effective to preclude further degradation and subsequent inoperability of the Appendix R fire wrap. At the conclusion of this inspection period, the VY staff was completing repairs to the fire wrap and continuing the root cause evaluation initiated via Event Report 95-0443 for this issue. The inspectors continue to monitor VY staff actions in the area of fire protection and Appendix R compliance (reference Sections 4.2 and 5.4 of this report).

3.1.2 Service Water Pin Hole Leaks

On August 18, an AO identified water dripping from the SW piping supplying the "B" emergency diesel generator. Because this SW pipe was insulated, the exact source of the water could not be determined at the time of identification. Subsequently, the pipe insulation was removed and two through-wall pin hole leaks were found.

The leaks were located at the 5 o'clock position on the 8 inch, carbon steel, seamless, schedule 40 pipe about midway along a straight 45 foot section of piping. The pipe location was free of pipe fittings and welds. The leaks were within two inches of each other and subjected to approximately 90 psig system pressure.

Vermont Yankee appropriately characterized the through-wall pin holes as a degraded condition outside the applicable piping design code. In addition, the effect of the leakage on alternate cooling tower (ACT) deep basin inventory required evaluation to determine whether this inventory loss was adverse to ACT operability. Based on this through-wall pipe leak and the un-analyzed ACT inventory loss, VY declared the ACT and "B" SW subsystems inoperable at 9:30 a.m. These systems being inoperable placed the plant in a 24-hour shutdown LCO. The NRC operations center was promptly notified in accordance with 10 CFR 50.72 of this TS required shutdown LCO (TS 3.5.D).

VY reviewed and implemented the guidance of Generic Letter 90-05, Temporary Non-Code Repair of ASME Class 1, 2, and 3 Piping, for their SW pipe through-wall leak evaluation and structural assessment. A flooding and spray evaluation was also conducted. The inspector noted that Yankee Nuclear Services Division (YNSD) assisted in the through-wall evaluation and piping structural integrity evaluation. The results were discussed with the NRC staff and verbal relief from ASME Class 3 permanent piping repair requirements was obtained on August 18, and VY exited the 24-hour shutdown LCO. Based on the ultra-sonic test results, the wall thinning did not represent a personnel or equipment safety concern. The inspector observed the ASME Level III inspector conduct a calibration and linearity check of the ultrasonic testing (UT) equipment and verified the quality assurance of the gauge block. As stipulated in GL 90-05, VY plans to conduct periodic leak assessments to verify structural integrity until permanent repair during the next plant shutdown of sufficient duration. Vermont Yankee also checked a variety of susceptible locations in that system and found no additional problems. On August 24, a rubber-gasket pipe clamp was installed to minimize SW leakage and spray.

In summary, VY appropriately implemented the guidance of Generic Letter 90-05. VY management was promptly informed of this degraded condition and assigned engineering and management oversight to assure appropriate and timely resolution of the problem. The scope of the ultrasonic testing provided assurance that the degradation was properly characterized. Thorough evaluations for flooding, spray, fire protection, and EQ were conducted to support the operability determination. VY's safety judgements were based on sound engineering evaluations.

3.1.3 LCO Maintenance

The inspector reviewed the work packages for planned maintenance on the SLC system, the Vernon Tie line, and the diesel driven fire pump. These activities were planned in advance of the preventive maintenance, received appropriate management reviews, and were conducted within the time period established. The inspector noted that appropriate entries into TS LCOs were made when the systems were removed from service and that CROs were knowledgeable of the work scope. The inspector also verified that flow rate and discharge pressure acceptance criteria in the urveillance procedure were based on TS performance requirements.

The conduct of the above planned maintenance activities was well controlled. Changes to the work scope and related work activities were properly reviewed and assessed against the scheduled outage window. Prior to the removal of these systems from service, the inspector reviewed the work packages and noted that VY had reasonable justification to remove the systems from service. No unacceptable increases in system unavailability for maintenance occurred.

3.2 Surveillance (61726)

The inspector reviewed procedures, witnessed testing in-progress, and reviewed completed surveillance record packages. The inspector observed that all tests were performed by qualified and knowledgeable personnel and conducted in accordance with VY TS.

3.2.1 System Performance Monitoring

Vermont Yankee conducted performance monitoring of the reactor building closed cooling water (RBCCW) heat exchangers (HX) to evaluate system operating parameters. As previously docketed, the SW supply piping and associated valves to the RBCCW HXs were modified to improve SW flow control and to reduce RBCCW and SW flow-induced vibration. Components within this system have failed and/or degraded due, in part, to SW flow conditions. The vibration, flow, and differential pressure (dp) monitoring conducted (since startup from the 1995 outage) were performed to verify and validate actual system performance against design expectations as defined in Engineering Design Change (EDC) 94-410. The system performance monitoring was conducted by YNSD and Operations Department personnel. The SW flow to the HXs was throttled to establish systems conditions necessary for performance monitoring. The RBCCW heat loads are not safety related by the SW portion interfaces with the safety related portion of SW.

The inspector observed the conduct of this performance monitoring and noted personnel performance and test control weaknesses as described below:

Neither the AO nor the test engineer recognized that the "A" RBCCW HX dp were exceeding the procedural limit of 10 psid when the HX dp instruments were un-isolated and placed in service at the onset of the performance monitoring. The "A&B" HX dps were approximately 21 psid and 12 psid, respectively. The test was immediately halted, pending the evaluation of this condition. The subsequent VY engineering evaluation confirmed that the increased dp was not adverse to HX structural integrity and confirmed that the limit could be increased to 25 psid. The applicable operating procedures were revised. The inspector noted that the shift supervisor directed the AO to operate the SW and RBCCW systems in accordance with their respective operating procedures.

These service water system manipulations for the purpose of performance monitoring were conducted in the absence of administrative control and without specific or special guidance or instructions. The inspector noted that the cognizant engineer was directing the manipulation of the SW flow to satisfy the EDC 94-410 performance monitoring commitment. However, no procedural instructions were evident in the field. The system operating procedures are classified by VY as "reference use," and therefore did not require "in-hand" continuous use. This particular system performance monitoring was scheduled and discussed (in general terms) during the CR pre-shift brief. However, neither the SS nor the operations planning group (OPG) knew the full scope of the monitoring. This was evidenced by the installation of the temporary flow monitoring instruments. The inspector noted that the HX performance monitoring was not discussed during the Plant Manager morning meeting.

In summary, the RBCCW HX performance monitoring was poor as evidenced by the lack of controls for the proper implementation of this activity. Instructions for the conduct of this performance monitoring were not effective. The use of the "reference use" system operating procedures as guidance for the AO in conducting this performance monitoring lacked specificity. In this case, system manipulations for the purpose of data taking challenged the defined operating envelop for system performance optimization. The cognizant engineer's guidance consisted of the EDC commitment item and a miscellaneous data sheet which lacked an assurance of a quality work input or output. Plant staff reviews and approval of this activity were minimal. VY management indicated they would review the adequacy of controls for the conduct of system manipulations for performance monitoring and/or data taking. This issue is unresolved pending further VY and NRC staff review **(95-19-01)**.

3.2.2 Safety Relief Valve Bellows Pressure Switch Failure

On July 25, while conducting surveillance testing in accordance with RP-4386, Main Stem Line Relief Valve Bellows Leakage Instrument Calibration/Relief Valve Bonnet/Sense Line Venting, the "B" safety relief valve (SRV) bellows pressure switch failed. This pressure switch failure required the CROs to declare the "B" SRV inoperable, enter TS 3.6.D.1, and reduce reactor power to below 95 percent until SRV operability could be restored. The inspector verified that the CR staff took appropriate and timely action to reduce reactor power below 95 percent. The instrumentation and control staff subsequently replaced the failed pressure switch and completed the surveillance test satisfactorily. Control room operators declared the "B" SRV operable and returned reactor power to 100 percent. The inspector concluded that operator and plant staff response to this problem was good.

4.0 ENGINEERING (37551, 92903)

4.1 Inservice Testing Deficiencies

On July 12, the VY engineering staff concluded that two residual heat removal (RHR) system keep fill check valves and two core spray (CS) system keep fill check valves (RHR-36A, RHR-36B, CS-33A, and CS-33B, respectively) were not being individually tested in accordance with ASME/ANSI OMa-1988, Part 10, "Inservice Testing of Valves in Light-Water Reactor Power Plants". This conclusion was based upon an internal review of NUREG 1482, "Guidance for Inservice Testing of Nuclear Power Plants", by the engineering staff which identified that these four check valves have a common upstream check valve (V64-20) in the condensate demineralizer system. The existence of check valve V64-20 and the current check valve IST test methodology prohibits conclusive verification of reverse flow closure by each RHR and CS check valve. The applicable procedures (OP-4123 and OP-4124) tested the function of these check valves as a pair. That is, both check valves (the RHR or CS check valve of interest and V64-20) would have had to fail to close to detect a problem. The inspector notes that the quarterly inservice test verifies valve closure by cessation of flow as measured by pump differential pressure.

As a result of the above determination, the VY staff completed a subsequent evaluation which concluded that there was reasonable expectation that valves RHR-36A, RHR-36B, CS-33A, and CS-33B were operable. The inspector verified that this operability determination was consistent with the guidance of Generic Letter 91-18 and NUREG-1482. The inspector noted that on July 13 the licensee conducted radiography of these four valves to verify their closure function and identified that each valve was properly closing under reverse flow conditions. This non-intrusive examination was permitted by ASME/ANSI OMa-1988, Part 10, paragraph 4.3.2.4, "Valve Obturator Movement".

The inspector reviewed the radiographs and based upon discussions with the responsible engineer determined that radiography will be performed quarterly on the subject check valves, coincident with the performance of OP-4123 and OP-4124. The inspector also determined that a system modification is being considered to allow individual valve testing.

The inspector noted the problem to be self-identified and concluded that the VY staff has taken appropriate actions. Accordingly, this IST deficiency (improper testing of safety related check valves) constitutes a violation (TS 4.6.E.1 and 2 and NRC approved IST Program) of minor significance and is being treated as a non-cited violation, consistent with Section IV of the NRC's enforcement policy.

4.2 Appendix R Programmatic Deficiencies

On July 25, VY reported that design deficiencies existed in the implementation of the Appendix R hot short mitigation strategy. Specifically, 24 MOVs required to achieve safe shutdown were susceptible to hot short failure (reference NRC Information Notice 92-18). On July 27, VY determined that safety relief valve cabling was also vulnerable to fire-related damage and therefore could not be credited for one fire scenario within RB zone 3 (RB-3). On August 1, RCIC cabling in RB-3 for the inside containment steam isolation valve was also found to be vulnerable to a fire and potentially incapable of fulfilling its Appendix R function. On August 23, VY found that the HPCI and RCIC high energy line break (HELB) temperature isolation circuits in RB-2 would potentially fail during a particular RB fire scenario, thereby causing the inadvertent isolation of these high pressure injection sources. Compensatory fire watches were established to identify, report, and protect Appendix R systems from fire damage.

On August 23, VY identified that their safe shutdown timeline of 43 minutes to commence reactor vessel water injection prior to core uncovery was non-conservative by 18 minutes, assuming all applicable Appendix R assumptions. The new 25 minute injection requirement is also less than the revised (and current) core reflood time of 28.5 and 30 minutes. At the conclusion of the inspection period, VY preliminary calculations showed that level will decrease to 20 to 30 inches below top of active fuel and that fuel clad temperature would slightly increase prior to injection. To preclude this specific problem, VY took credit for the above mentioned fire watches and added another fire watch to the west switchgear room. VY continued to assess whether procedural, personnel, or equipment changes could be implemented to obtain core injection in 25 minutes.

Based on VY information, the RCIC system is potentially incapable of performing its Appendix R function and VY would not be able to inject water prior to core uncovery. In addition, if an Appendix R fire were to occur in RB-3 and using Appendix R assumptions, the reactor would remain at pressure cycling on the safety valves while reactor water level slowly decreases. The inspector notes that Appendix R assumes the immediate and concurrent loss of all equipment within the area subject to an Appendix R fire. The safety significance of the above stated problems is low based upon the fire protection/prevention measures currently in place and the consequently low probability of a plant fire. However, these patterns represent a weakness in VY's implementation of Appendix R requirements.

The inspectors continued to monitor VY's identification and resolution of Appendix R deficiencies. Appropriate management oversight and safety reviews have been conducted to date. Independent and technical expertise has been dedicated to review, assess, and correct the identified problems. An overall integrated self-assessment will also be performed. A Vermont Yankee letter dated August 22, 1995, summarized the licensee's activities to date. The NRC fire protection staff has reviewed the problems and licensee corrective actions; no immediate plant safety concerns have been identified.

4.3 Service Water Pipe Temporary Modification

On August 23, the Plant Operations Safety Review Committee (PORC) reviewed and recommended to the Plant Manager that a SW pipe clamp be installed per Temporary Modification 95-047. The temporary modification (TM) described the use of two pipe clamps with rubber gasket material to cover the two pin hole leaks.

On August 25, the inspector performed a field inspection of the TM and noted that the clamp configuration was not that approved by PORC. Specifically, a single pipe clamp configuration was used vice a double pipe clamp. This was of some importance due to potential changes in pipe loading and stress assumptions. In addition, the inspector noted that the TM was installed and signed off as satisfactorily completed, despite the fact that clamp leakage exceeded the no leakage TM installation criteria. These and other observations involving TM instructions were discussed with the Maintenance Manager and appropriately resolved.

The inspector concluded that the pipe clamp installed per TM 95-047 was adequately implemented. The TM instructions were unclear as to torque and structural verification requirements and the installation configuration changes were made without appropriate consideration of TM change requirements. However, the pipe clamp installed was of sound design, met the intent of the TM, and did not adversely effect pipe wall integrity.

4.4 (Closed) TI 2515/128: Plant Hardware Modifications to Reactor Vessel Water Level Instrumentation

This inspection was to verify and evaluate certain aspects of the modification made to the reactor vessel water level instrumentation that was made by VY in response to NRC Bulletin 93-03. Various aspects of this modification had previously been reviewed by the NRC (reference NRC inspection reports 93-21 and 94-10. The focus of this inspection included VY's actions on information concerning a postulated transient initiated by closing one of the manual valves in the instrument system, isolating the instrumentation reference leg from the reactor vessel. This issue was discussed within NRC Information Notice (IN) 93-89, Potential Problems with BWR Level Instrumentation Backfill Modifications, dated November 26, 1993.

The inspector based this review upon information in plant design change (EDCR) 93-404, its 10 CFR 50.59 safety evaluation and supporting documents, plant procedures and records. Specifically, the inspector verified that VY had established a backfill flow rate that was based on engineering calculations. Those calculations (VYC-1199, VYC-332 and VYC-1276) established an acceptable flow rate range that was sufficient to prevent the migration of non-condensable gas into the reference leg piping, but less than the flow rate that would create a thermal stress gradient at the reactor vessel nozzle weld, or would result in a pressure drop within the instrument reference legs sufficient to affect the reactor vessel indicated level. The safety evaluation properly considered the potential for single or common mode failure for system flow being lost or in excess. Also a review was completed for impact on Emergency Operating Procedures (EOP) actions. Plant procedure OP-2111, Control Rod Drive System, requires a normal flow range for back fill of between 0.001 and 0.005 gallons per minute. These flow rates are verified by an operator daily in accordance with AP-0150, Conduct of Operations and Operator Rounds. The inspector observed that all four flow instruments were indicating within this range.

OP-2111 also establishes procedural requirements for placing the backfill system in service and removing from service. Vermont Yankee has established administrative limitations (OP-2111, section 2.g) for the backfill being out of service up to fourteen days. The procedure requires that a basis to maintain operation evaluation be performed if the system is out of service beyond this limitation.

The information within NRC IN 93-89 was evaluated by Yankee Atomic for VY and the results transmitted in memorandum file Nos. E93-404\MEMO-11M and E93-404\MEMO-12M, dated December 8, 1993 and March 17, 1994, respectively. Vermont Yankee concluded that the backfill modification did not change the likelihood or severity of a transient caused by isolating a reference leg. Their position is that the potential to isolate the reference leg, always existed. However, with the backfill system in operation, the sensors respond immediately, where a slower response would occur if the system was not in operation. The design change, EDCR 93-404, did not include analysis of the IN 93-89 issue within its 10 CFR 50.59 Safety Evaluation as it preceded promulgation of the information within IN 93-89. The design change did, however, recognize that the drywell reference leg isolation valve could be closed and that this action would pressurize the piping to the Control Rod Drive charging header pressure. As part of the modification, the design pressure specification of the reference leg piping affected by this isolated condition was changed and the piping was hydrostatically tested.

Vermont Yankee has also changed the status of the four reference leg containment isolation valves to "locked open" within the valve alignment list, Appendix C of OP-2115, Primary Containment. To ensure continued compliance with the specified alignment, VY performs a quarterly surveillance of locked valves in accordance with AP-0155, Current System Valve and Breaker Lineup and Identification and VY form 0155.04. The four reference leg isolation valves, RV-12A and B, and RV-18A and B are included in that list of locked valves. The inspector verified that the required locking devices were installed on these valves.

Vermont Yankee has evaluated and tested the modified system as part of the design change activities. These tests measured vessel level performance with changes in backfill flow. NRC Inspection report 94-10, section 3.0 addressed the NRC evaluation of these analyses and tests.

The reactor vessel pressure instrument lines outside of the primary containment are classified as safety class 2, with the class change being a restricting orifice that is in each line and is installed in the primary containment portion of the piping. Each line has a normally open manual isolation valve and excess flow check valve pair located at the containment penetration. Those valves are functionally tested during refueling outages in accordance with OP-4378, Excess Flow Check Valve Functional. The backfill modification injects water from the control rod drive system on the instrument side on the excess flow check valves.

Vermont Yankee has established a safety class break of the modified system at the control rod drive system side of the two check valves through which injection flow passes into the instrumentation piping. This is the boundary between safety class 2 and non-nuclear safety piping, and is also the seismic class 1 boundary. Vermont Yankee provided continuation of the divisional separation of the reactor instrumentation by placing each pair of check valves on physically separated racks. These check valves were designated as containment isolation valves and leak tests were developed.

As part of the modification process, OP-4030, Type B and C Containment Leak-Rate Testing, was revised to establish reverse direction leak rate tests for the newly installed check valves (2-3-430A/B, -432A/B, -433A/B, and -435A/B). Completion of these leak rate tests was a prerequisite of the design change installation and test procedure. Initial tests of these eight newly installed valves revealed that one valve, 2-3-430B, had an unacceptably high leakage rate of 0.195 pounds mass per hour on October 14, 1993. The valve was replaced and tested satisfactorily on October 14, 1993 before system testing continued.

Vermont Yankee had revised the inservice test program plan to include these eight check valves in Table 5-1, drawing G-191267, sheet 2. The valves have been designated category A/C, with required leakage rate and full-stroke tests in the closed direction. A refueling outage test frequency justification, ROJ-V15, was in place to establish that test frequency.

During the earlier review of this modification in NRC inspection report 94-10, weaknesses were observed in supporting documentation involving thermal stress calculations for the case where an accumulation of noncondensable gases in the condensing chamber could result in reduced steam transfer from the reactor vessel to the condensing chamber thereby reducing the rate of condensate flow and reducing temperature in the condensing chamber. This scenario would result in cooler water flowing back to the reactor vessel. The original calculation only assessed the thermal effects on the nozzles and did not evaluate the potential effects on the condensing chamber or its associated piping.

In response to this issue, Yankee Atomic performed calculation VYC-1276, Heat Transfer Analysis in Condensate Chamber Return Pipe. It was generated to estimate the temperature of the liquid as it flows back to the reactor vessel nozzle via the return pipe of the condensing chamber, in the event that the chamber was filled with non-condensable gasses. This calculation was reviewed and approved by April 13, 1994. Yankee Atomic concluded that the analysis of record for the structural integrity of the reference leg piping, VYC-493, uses thermal conditions that envelope those used in VYC-1276. The worst case thermal stresses for this condition were within those used to qualify the piping to the ANSI B31.1, 1977 Edition of the piping code.

The inspector also discovered during inspection 94-10 that VY had not established any periodic calibration or functional tests of the system flow indicators at that time. In response, VY had modified procedure OP-5371, Instrument and Control Routine Outage Activities, on October 27, 1994, to add a section P, Reference Leg Keep Fill Flow Calibration. This activity is a functional test of the four flow monitors that is performed by establishing a flow rate of 0.0050 gallons per minute for 15 minutes and 51 seconds. The captured water should be within 50 ml of 300 ml.

The inspector found that VY had effectively evaluated the issues concerning the reactor vessel level instrumentation design modification, including those that were stated in NRC IN 93-89. Vermont Yankee also acted on the NRC inspection findings stated in NRC inspection reports 93-21 and 94-10. This completes the actions for TI 2515/128.

4.5 (Closed) VIO 94-04-01: Test Program for the Control Room Ventilation System and LER 94-17: Control Room HVAC Damper Failure

An finding from NRC inspection report 94-04 identified that VY had failed to establish a test program that demonstrates that the control room ventilation system would perform its safety function. In response, VY modified the heating, ventilation and air conditioning (HVAC) surveillance procedure, OP-4192, to include testing of control room ventilation system components. During separate tests conducted on December 10, 1994 and January 17, 1995, the control room fresh air inlet damper failed to close. The first event occurred during quarterly testing, the second while performing maintenance inspections. Both of these equipment failure events were addressed in Licensee Event Report (LER) 94-17, dated January 26, 1995. The status of ventilation system preventive maintenance was reviewed in light of these damper failures during NRC Inspection 95-80. Section 2.2 of that report detailed findings relating to VY handling of corrective actions for HVAC deficiencies.

The inspector reviewed OP-4192 and confirmed that the isolation test for the ventilation system was detailed in section 4192.05. Records indicated that VY was conducting the test, as required. However as detailed below in the discussion of the VY root cause analysis for LER 94-17, the test was not fully appropriate to verify the system isolation safety function.

Vermont Yankee made a root cause evaluation of the ventilation damper failures addressed in LER 94-17. It found that the surveillance test for ventilation isolation was ineffective in determining damper seal integrity. The analysis also identified improper safety classification of the component damper, a communications breakdown between plant departments, improper lubrication of the failed damper, modifications for improved access and an improved damper configuration. The root cause analysis was thorough and appropriate and provided valuable recommendations for improvement. Most have been implemented.

In addition to upgrading the surveillance procedure, VY personnel made an evaluation of industry practices relating to benchmarking control room ventilation system test practices. Although the survey documented the wide difference in system designs and licensee testing requirements, it did result in several specific recommendations being made for improvement in VY's test and preventive maintenance practices.

Vermont Yankee also conducted a survey of safety class systems with the intention of identifying any additional testing. "Safety class" systems considered in this area were those either directly or indirectly relied upon for operability within the VY TS, the Standard Technical Specifications (NUREG-1433) and the VY Individual Plant Examination. A systems matrix was used to develop recommendations for test changes of each analyzed system.

The inspector reviewed the results of the bench marking survey and of the safety class system test study. The prioritized recommendations appeared of value and appropriate. Both are comprehensive. The industry benchmark survey assisted VY in establishing an effective test interval. The safety class system test study identified recommendations to enhance testing within 28 systems or sub-systems. Those recommendations are being studied by the responsible VY department for resolution.

The corrective actions documented in the March 18, 1994 response to the notice of violation have been completed. The short term corrective actions stated in LER 94-17 are likewise completed. The long term corrective actions from LER 94-17, have also been completed with the exception of the upgraded HVAC preventive maintenance program. This outstanding action is being tracked by VY (LER 94-17-01). This violation and associated LER are closed.

5.0 PLANT SUPPORT (71750)

5.1 Radiological Controls

5.1.1 Routine Radiological Observations

Corrective and preventive maintenance was conducted on a RB forced-air room cooler (RRU 13). The inspector verified that radiation and contamination surveys were conducted prior to the maintenance and following pipe insulation removal. Workers were cognizant of radiation protection requirements and articulated good knowledge of radiological work conditions.

As documented in Inspection Report 95-17 and 95-18, the controls associated with locked high radiation doors were identified as inadequate to preclude unauthorized entry and corrective actions were implemented to improve performance in this area. This period, the inspectors noted effective control of locked high radiation doors. In particular, an appropriate sensitivity for the proper control of high radiation doors was demonstrated when an AO identified that a locked high radiation door key was missing. Immediate corrective actions included positive verification that the door was locked, the installation of a new lock to assure positive control, and a comprehensive search to locate the missing key. An Event Report was initiated and management reviews were commenced to preclude recurrence.

5.1.2 ALARA Review for Appendix R Fire Watches

During a routine plant tour of the RB on August 8, the inspector noted that the continuous fire watch (Section 4.2), established on the 252 foot level of the RB was positioned in a location previously identified as having dose rates higher than other areas within the RB. In this particular area, near the north hydraulic control units beneath "A" RHR piping, VY had previously determined that security officers posted for refueling outage compensatory measures were accumulating radiation exposure not representative of As Low As Reasonably Achievable (ALARA) considerations. The inspector also noted that a radiation work permit (RWP) was not established specifically for fire watch duties. The inspector informed the RP Staff of these observations and the continuous fire watch location was moved to a lower dose rate area that did not compromise the effectiveness of the fire watch. A specific RWP was also implemented for the fire watches. The inspector determined that subsequent to his questions being relayed to the VY staff, the VY ALARA engineer performed an ALARA assessment.

Overall, the inspector concluded that adequate controls were implemented to maintain personnel radiation exposures ALARA. Although the continuous fire watch location was moved subsequent to the inspector making his observations, personnel radiation exposures were reasonably low due, in part, to the generally low background radiation levels.

5.2 Emergency Preparedness Drill

On August 9, the VY staff conducted a planned emergency preparedness (EP) drill with participation by the Vermont, New Hampshire, and Massachusetts state and local emergency response organizations. The drill scenario progressed through the four emergency classifications to a General Emergency which necessitated the simulation of offsite shelter and evacuation protective action recommendations. The inspector observed various drill activities in the Technical Support Center (TSC) and the Operations Support Center (OSC).

The inspector observed generally good communications and coordination of resources to mitigate the consequences of the scenario accident conditions. An actual problem involving the failure of normal telephone communications between the simulator control room, TSC, and OSC occurred and was adequately compensated for by the VY staff. The drill critique conducted in the TSC immediately following the exercise was observed to be thorough and self critical.

5.3 Feedwater Copper Concentration

On July 13, an ER was initiated documenting an increasing trend in feedwater copper concentration. VY identified that since March 1995, copper concentrations were higher than normal (1.0 to 1.52 ppb) exceeding the fuel vendor water purity limit (1.0 ppb). Actions were accelerated to repair problems associated with the RWCU and condensate demineralizers and to assess the safety significance of the high copper concentrations. A potential contributor to the high copper was increased copper solubility and decreased resin efficiency due to warm weather conditions (Section 2.8). VY has established administrative limits based upon industry information and periodically analyzes for copper. Copper is not a TS required reactor coolant chemistry parameter. However, high reactor coolant copper concentration may adversely impact reactor fuel integrity by contributing to crud build-up, localized corrosion, and fuel clad heat transfer degradation.

The inspector noted that unlike previous good trend assessment of changing reactor coolant chemistry, VY management was unaware of the increasing trend until the feedwater copper concentration limit was exceeded (limit established by their nuclear fuel supplier, General Electric Company). As detailed in the vendor report, VY exceeded the "continuous" water purity control limit during a fourteen day period.

The inspector discussed this issue with the Chemistry Manager who stated that although exceeding the vendor water purity limits was of concern, it did not represent a significant reactor safety issue. The vendor limits were established to provide a reasonable warranty condition for reactor fuel. Notwithstanding, prolonged exposure of fuel bundles to poor water chemistry conditions does increase the potential for fuel pin cladding degradation and a potential longer term radiological concern.

5.4 Fire Watch Instructions

On about August 3, one week after the establishment of the fire watches to compensate for Appendix R safe shutdown deficiencies, a Shift Engineer (SE) identified that the roving fire watches were not touring all required areas in RB zone 3 (reference Section 4.2). VY management was immediately informed of this occurrence, initiated an ER, enhanced the fire watch instructions, and briefed fire watches as to the required tour areas and the equipment cabling of concern. The inspector reviewed the fire watch instructions, interviewed fire watches, and determined that the immediate corrective actions provided adequate assurance that the fire watches would be properly conducted.

The inspector reviewed plant procedure OP-0042, Fire Protection Program, and noted that the responsibility to write fire watch instructions, implement, and manage fire watches resides with the SE. The fire watch instructions are generally based on information provided by the persons and/or plant organization requesting the fire watch permit. For this particular case, the Electrical Engineering and Construction (EE&C) Department provided the technical basis and reasons why this particular Appendix R fire watch was required to compensate for identified Appendix R deficiencies. The inspector noted that combustible material control, fire barrier surveillance and assessment, and the Fire Protection Program ownership (in general) resides in the Technical Programs Department. In addition, a fourth organization (YNSD) does a significant portion of the VY Safe Shutdown Capability Analysis and Fire Hazards Analysis; two key documents providing the basis for VY's fire protection strategy.

The inspector reviewed the applicable fire watch instructions and EE&C information, and determined that adequate information was provided to the Operations Department staff for the correct implementation of fire watch instructions. Drawings and verbal descriptions of the RB zones and particular Appendix R deficiency were provided to the Operations staff. Amplifying information regarding the systems and components of concern were also provided.

The inspector concluded that the Operations staff did not effectively translate the engineering information into clear and detailed fire watch requirements. This contributed to the failure to properly implement one of the many compensatory fire watches required to compensate for Appendix R programmatic and hardware deficiencies. This failure constitutes a violation of minor significance and is being treated as a non-cited violation, (10 CFR Appendix R, B.3.g) consistent with Section IV of the NRC's enforcement policy.

6.0 EXIT MEETING

Meetings were held periodically with VY management during this inspection to discuss inspection findings. A summary of preliminary findings was also discussed at the conclusion of the inspection and prior to report issuance. No proprietary information was identified as being included in this report.