U.S. NUCLEAR REGULATORY COMMISSION
REGION I

`eport No. 92-06

Docket No. 50-271

Licensee No. DPR-28

Licensee: Vermont Yankee Nuclear Power Corporation
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Facility: Vermont Yankee Nuclear Power Station
Vernon, Vermont

Inspection Period: March 7 - April 21, 1992

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Inspection Summary: This inspection report documents resident safety inspections conducted between March 7, 1992 and April 21, 1992. Station activities inspected during this period included: plant operations; radiological controls; maintenance and surveillance; emergency preparedness; security; engineering and technical support; and safety assessment and quality verification.

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

EXECUTIVE SUMMARY
Vermont Yankee Nuclear Power Station
Report No. 50-271/92-06

Plant Operations

The conduct of operations during the refueling and maintenance outages, and shutdown and startup activities, were well controlled and contributed to the safe operation of the plant. Five activities resulted in the inadvertent actuation of Engineered Safety Features that challenged control room operators. However, these events were of minor safety significance and no programmatic concerns were identified. The operating organization's initial response to concerns involving reactor mode switch integrity reflected a strong safety ethic. A walkdown of the core spray system identified no conditions that would affect system operability. Housekeeping was generally good, however the cleanliness of the drywell did not meet Vermont Yankee management expectations. Refueling and spent fuel inspection activities were well controlled.

Radiological Controls

A review of radiological event response and radiological housekeeping identified no areas of concern. An inspection of Vermont Yankee's Respiratory Protection Program identified a few concerns regarding the training of technicians performing respirator maintenance. Recent changes in radiation protection supervision of radiological work contributed to better control of contamination, however, concerns were identified involving a lack of communication regarding personnel contamination information. Also identified was an issue where workers were exposed to higher than necessary radiation levels, but prompt action by the radiation protection department minimized the effect of this condition.

Maintenance and Surveillance

Improvements in equipment and tooling increased personnel safety and reduced personnel exposure. Very good maintenance and supervisory oversight contributed to Vermont Yankee's understanding of equipment failure mechanisms and the repair efforts necessary to restore the "B" emergency diesel generator to service. Direction provided by Operations Department management during the emergency core cooling integrated test maintained a high level of control and assured the safe conduct of this test. Corrective actions involving a missed Technical Specification surveillance requirement were appropriate and the issue was dispositioned as a non-cited violation.

Security

For Cause testing identified a number of employees not in conformance with the licensee's Fitness-For-Duty program. Corrective actions were good, and an issue involving completeness of For Cause testing was dispositioned as a non-cited violation. Vermont Yankee access control for the drywell was determined to be in accordance with regulatory requirements, and good control was maintained for material brought into the containment. The NRC notified Vermont Yankee that a change to their security plan constituted a decrease in plan effectiveness, although subsequent actions to resolve this issue were acceptable. The use of overtime by security personnel was found to be excessive, and additional management attention is warranted in this area.

Engineering and Technical Support

Corrective actions involving reactor vessel cladding indications and voltage setpoint drift in safety-related under-voltage relays were determined to be appropriate. Vermont Yankee's evaluation and actions to resolve reactor mode switch integrity concerns demonstrated a strong orientation toward nuclear safety.

Safety Assessment and Quality Verification

Plant tours conducted by all levels of plant management and supervision were effective in assuring the safe and proper conduct of activities during the outage. Guidance provided to control room operators regarding the loss of decay heat removal was determined to be good, however, a reluctance to formalize the guidance as a procedure was noted. A review of quality control documents associated with the repair of the "B" emergency diesel generator identified no concerns. Activities to resolve a residual heat removal service water leak were well controlled. An inspection involving Temporary Instruction 2515/113, "Reliable Decay Heat Removal During Outages," found that the licensee has incorporated industry guidance into their outage planning and management programs.

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DETAILS

1.0 SUMMARY OF FACILITY ACTIVITIES

On April 21 at 10:08 a.m., Vermont Yankee (VY or the plant) synchronized the main generator to New England power grid and safely completed Refueling Outage (RO) XVI. This milestone marked the completion of a 46 day outage that was entered on March 7 to begin a period of shutdown operations to support refueling, maintenance, and surveillance. RO XVI experienced emerging work and the accomplishment of maintenance activities associated with the unsatisfactory performance of various tests. All tests were subsequently performed satisfactorily and contributed to the assurance that the plant can operate safely. The major maintenance accomplished during this period included a main transformer replacement, electrical and emergency core cooling system maintenance, replacement of two low pressure feed water heaters, extensive erosion/corrosion inspections, motor operated valve testing, and high and low pressure turbine work. Selected reactor core or vessel maintenance and inspection items were accomplished, such as: the rebuilding of 12 control rod drive (CRD) mechanisms, a one-time exchange of CRD bolting, refueling of approximately one third the reactor core, and in-vessel inspections.

The high level of maintenance and surveillance associated with this outage contributed to five plant events that challenged control room operators. These events, documented in the Operations section of this report, were , with the exception of reactor mode switch concerns, of minor safety significance. Control room operators promptly assessed the safety system actuations, and initiated corrective action. There were no radioactive releases to the environment during this outage.

2.0 PLANT OPERATIONS (71707, 93702, 93700, 71710)

2.1 Inspection Activities

The inspector verified that the facility was operated safely and in conformance with regulatory requirements. Management control was evaluated by direct observation of activities, tours of the facility, interviews and discussions with personnel, and independent verification. The inspector performed backshift inspections including backshift and weekend inspections during this inspection period.

2.2 Significant Plant Events

2.2.1 Primary Containment Isolation System (PCIS) Group I Isolation and Reactor Scram During a Controlled Shutdown

At 7:46 p.m. on March 6, during the plant shutdown to enter RO XVI, a PCIS Group I isolation and a reactor scram occurred. Reactor power was less than one percent of rated. At the time of the event, the reactor mode switch (RMS) was in the "Startup" position, and should have precluded the less than 800 psig Group I isolation condition and the main steam isolation valve (MSIV) less than full open reactor scram condition.

Operators responded to the above condition in accordance with recovery procedures. Timely assessment and recovery actions by the operators occurred. Within three minutes of initiating condition, the MSIVs were reopened and reactor core decay heat removal was again provided by the plant's main condenser. The 4-hour 10 CFR 50.72 (b)(2)(ii) required report to the NRC was made in a very timely manner. All plant systems responded to the PCIS and reactor scram signals in accordance with their design. A comprehensive post-trip report was developed by the Operations Department as a result of this event. Subsequent investigation by VY determined that this event was caused by the RMS contacts not fully engaging in their proper location. VY analysis of this event was provided in their April 4 submittal of Licensee Event Report (LER) No. 92-05. The NRC's review of the VY evaluation of RMS concerns is contained in Section 7.3.

Since subsequent RO activities were RMS position dependent, and until a thorough investigation could be completed, VY instituted on March 7 an appropriate compensatory requirement to be implemented by the Instrument and Control (I&C) Department. This action was the result of a Plant Operations Review Committee (PORC) recommendation, and required that before any activity is initiated, based on a RMS position change, that all contacts associated with that position be verified. Night and Standing Orders, as well as I&C RMS contact verification forms, were developed by VY as appropriate administrative and work controls to provide proper implementation of the RMS integrity checks. No NRC concerns, as relating to the conduct of the verification activity, were identified.

The response to this event by the plant operating organization was excellent. The self-identification and timely implementation of the compensatory measure established by the PORC, until questions of RMS integrity could be fully addressed, demonstrated strong safety ethic.

2.2.2 Partial Reactor Scram Due to Personnel Error

On March 7 with the RMS in "Refuel" and all rods fully inserted, a partial reactor scram resulted from an I&C technician who inadvertently pulled fuses associated with the reactor protection system (RPS). This deenergized one of four RPS channels, causing one of four control rod groups (20 rods) to scram. The immediate actions to replace the fuses, reset the reactor scram, and make a timely event report to the NRC pursuant to 10 CFR 50.72 were appropriate. This event was of minor safety significance, because the plant was initially shutdown with all rods fully inserted and no subsequent rod movement occurred.

The scope of the maintenance activity was to pull the fuses associated with the turbine thrust bearing detector to allow removal of the main turbine generator's front standard. The switching and tagging order associated with the work request listed the fuses by their control wiring diagram (CWD) fuse identification. Because the technician failed to refer to the fuse selection verification document to determine the proper fuse identification on the panel, a MSIV fuse and a RPS fuse were pulled instead of those associated with turbine thrust bearing circuitry. Vermont Yankee uses the fuse selection verification document to cross-reference fuse designations on control panels to the fuse designations on the plant CWDs. Because the fuse designations often differ between the panels and the CWDs, the cross-reference is an often used document.

The I&C Department conducted department training on the use of the fuse selection verification document to ensure that all technicians fully understood its use and to reinforce initial I&C technician training given with regards to this document. The I&C Supervisor also discussed this event with the technician. The inspector considered these corrective actions to be appropriate and adequate to prevent recurrence. This event was reported as LER 92-06. The inspector had no further questions on this subject.

2.2.3 Inadvertent PCIS Group III Isolation Due to Improper Resetting of Refueling Floor Radiation Monitor

On March 8, a PCIS Group III isolation occurred during resetting of the refuel floor radiation monitors following removal of the reactor vessel steam dryer. All Group III isolation valves responded properly and the standby gas treatment system automatically started as required. The control room operators reset the isolation monitors and made a timely 4-hour event report to the NRC pursuant to 10 CFR 50.72. This event was of minor safety significance.

Vermont Yankee procedure OP 1200, Rev. 15, "Preparation of the Reactor Vessel for Refueling," states that a Group III isolation/Engineering Safety Feature (ESF) actuation can be expected during the removal of the steam dryer. Subsequent procedure steps are intended to prevent the actuation of a Group III isolation. The procedure requires that permission be obtained from the Shift Supervisor (SS) prior to bypassing and restoring the refuel floor radiation monitors. In addition, an appendix of OP 1200 provides further instructions to operators on how to bypass and restore the radiation monitors. The appendix steps are independently verified by a second operator and reviewed by the SS. The procedure did not specifically require the operator to reset the monitor. Consequently, on March 8, the PCIS initiation was actuated when the radiation monitor was taken out of bypass with the trip signal present.

The inspector noted that VY's original intention was to prevent the actuation of a Group III isolation during this evolution. Despite this, the Group III isolation was unintentional and was not pre-planned. Vermont Yankee planned to review procedure OP 1200 prior to its next use in order to enhance the instructions associated with the restoration of the radiation monitor to service. In addition, VY will implement appropriate changes to procedure OP 1201, Rev. 16, "Assembly of the Reactor and Drywell Systems," prior to its use at the end of the outage period. The inspector had no further questions on this item. Vermont Yankee has designated this event as LER 92-07.

2.2.4 Inadvertent PCIS Group III Isolation Due to Improper Jumpering of a Relay Coil

On March 15 with the RMS in "Shutdown," an inadvertent actuation of one half of the PCIS Group III isolation system occurred during I&C maintenance on a MSIV relay coil. All affected Group III isolation valves responded properly and the standby gas treatment system automatically initiated. The I&C technician immediately recognized his error involving the installation of a jumper and informed the control room. The control room operators responded to the Group III isolation and made a timely 4-hour event report to the NRC pursuant to 10 CFR 50.72. This event was of minor safety significance.

The intent of the jumper was to allow removal of a relay coil while maintaining power to downstream relays. However, while jumpering the relay, the technician inadvertently touched the grounded jumper to the high voltage side of an adjacent relay circuit causing a high current condition that blew the circuit power supply fuse (5 amperes). This de-energized a portion of the PCIS Group III isolation circuitry.

The inspector discussed this event with the cognizant I&C engineer, inspected the area in which the work was performed, and concluded that this event was an isolated occurrence due to a personnel error. This event will receive further NRC inspection during the review of LER 92-08 submitted by VY on this subject.

2.2.5 Inadvertent PCIS Group IV and Group V Isolation Due to Inadequate Pre-Job Review

On March 31 at approximately 5:30 p.m., an I&C technician removed the power supply leads to protection circuit relays and unknowingly removed power to relays associated with the PCIS. This activated a Group IV isolation of the residual heat removal (RHR) system and a Group V isolation of the reactor water clean up (RWCU) system. When the isolations occurred, the RMS was in "Shutdown," the RWCU system was inoperable due to maintenance, and the "A" RHR system was operating in the shutdown cooling mode. All isolations actuated properly. The plant restored reactor vessel cooling and promptly made a 4-hour event notification to the NRC. This event was of minor safety significance, due to low reactor decay heat.

The work order (WO) issued to control this activity was independently reviewed by the technician and the job planner/foreman; however, the reviews were not of sufficient detail to identify all relays that would be de-energized while performing this work. Vermont Yankee determined that: (1) the responsible technician should have initiated VY lifted leads and jumpers requirements or should have more carefully reviewed the prints prior to initiating the work; and, (2) a more thorough independent review could have prevented this event.

The inspector discussed the details of the event with the involved technician, the job foreman, and with the planner responsible for the independent review and the job foreman, and concluded that VY's assessment of the cause of this event was accurate. The NRC will further review the event details and VY corrective actions during the review of LER 92-11.

2.2.6 Emergency Core Cooling System (ECCS) Actuation With Reactor Head Removed

On April 12 at 5:27 a.m., a spurious low low reactor vessel water level signal occurred due to improper restoration of reactor water level differential pressure (d/p) transmitters. This resulted in a reactor scram, ECCS initiation, and Group I through VI isolations. The event initiated while the RMS was in "Refuel" with the reactor head removed to support vessel head-to-flange o-ring replacement. Reactor vessel water was initially controlled at approximately 298 inches (five inches below the vessel flange). The level surge due to ECCS injection caused the indicated level to approach 308 inches. The subsequent water injection flow from the core spray system to the reactor vessel overflowed the reactor flange and drained into containment through open manway covers. Approximately 3000 gallons were collected by the radioactive waste treatment system and processed for re-use. Personnel inside the containment, at the time of ECCS initiation, promptly evacuated when they heard the core spray system actuate and water started to cascade down from the top of the containment. No personnel contamination or injuries were reported. There were no personnel in the vicinity of the reactor vessel flange. Preliminary inspections by operations personnel and the inspector of the containment area indicated that there was no identified damage.

A preliminary VY report indicated that a poorly coordinated effort to restore the level d/p instrumentation resulted in the reactor scram on the spurious low reactor water level (112 inches) signal and the ECCS initiation on low low level (82.5 inches). The level transmitter was returned to service following the maintenance on an excess flow check valve on the transmitter's instrument line. The I&C technician placed the level d/p transmitters in service prior to the Auxiliary Operator (AO) opening the instrument line root valves. This action was not known to the AO, because he subsequently restored the reference leg first, causing the level transmitter to sense a very low reactor vessel water level condition. This, in effect, resulted in a simulated low reactor water level condition, even though the reactor vessel water level was constant at 298 inches. Initiation signals were received by all ECCS systems; however, only the core spray systems injected into the core. RHR train "A" was isolated for maintenance and the "B" train was in the shutdown cooling mode. Operating RHR service water (RHRSW) pumps tripped as required. The alternate rod insertion/recirculation pump trip protective features initiated, however, the recirculation pumps were not operating. Both emergency diesel generators (EDGs) automatically started as required, but because the vital buses were continuously energized, the EDGs did not automatically close in to their respective buses. The PCIS Group I through VI isolations actuated properly. The inboard MSIVs were already shut for maintenance.

The inspector noted that during RO XV, similar maintenance on excess flow check valves resulted in several PCIS isolations. These isolations, documented in Inspection Report 90-15 and LER 90-18, were caused by procedural inadequacies. Previous corrective actions would not have reasonably prevented the April 12 event.

VY determined that this event met the 4-hour event notification reporting criteria of 10 CFR 50.72. A prompt and accurate report was made. The inspector will continue to review VY corrective actions as part of followup for the expected LER.

2.2.7 Fuel Pin Failure and Inspections

As a result of fuel inspections performed by General Electric (GE), fuel pin D-1 of assembly LYJ001 was identified to have a longitudinal clad failure approximately 12 inches long, 28 inches from the bottom, and 1/8 inch wide. Inspections concluded that all fuel pellets were intact. LYJ001 was a third cycle fuel assembly that under went two full cycles and one partial cycle of pre-conditioning interim operating management recommendations. During the third cycle, the assembly was 1 of 4 center fuel assemblies. Pin D-1 was an outer assembly pin located mid-faced on the control rod blade. The preliminary root cause was determined to be a tubing reduction flaw. Through discussions with the GE lead engineer performing the fuel inspections, the inspector was informed that a fuel pin crack of the size described above could result in the offgas levels experienced by VY during Cycle XV. The prior operating cycle offgas levels were described in Inspection Report 92-04.

The inspector observed a portion of the fuel inspections performed by contractor personnel and concluded that the inspections were in accordance with approved procedures. These inspections, intended to help assess fuel performance, consisted of: visual, fuel rod oxide thickness, and longitudinal growth measurements. Yankee Nuclear Services Division (YNSD) Quality Control inspectors also performed independent inspections of this activity. In addition, the inspector noted that the radiological, work control, and housekeeping practices employed by the contractors met VY procedural requirements. Frequent supervision and direction by VY radiation protection (RP) technicians maintained a high standard of work.

2.3 Operational Safety Inspections

2.3.1 ESF Walkdown

The inspector performed a complete walkdown of the accessible portions of the core spray system. The inspector confirmed that the line-up procedures matched piping and instrument drawing (P&ID), G-191168, Rev. 26, Flow Diagram - Core Spray System, and the as-built configuration. The system's maintenance and instrument calibration records were also reviewed, with no weaknesses noted. A Technical Specification (TS) required surveillance was observed and is discussed in Section 4.4.1. No conditions which might degrade core spray system performance were identified.

2.3.2 Drywell Inspection

On March 19, the inspector conducted a tour of the drywell to inspect activities in progress and perform a walkdown of the core spray system. The inspector noted that drywell cleanliness was weak, and expressed this concern to operations management who indicated awareness of the issue. Initiatives were underway to more aggressively deal with the issues of tool control, cleanliness, and workplace safety. Subsequently, on March 26 the conditions in the drywell were reviewed and the inspector noted improvements in cleanliness and housekeeping practices. The inspector had no further questions.

2.3.3 Housekeeping Observations

Overall, the housekeeping within the reactor building (RB) and turbine building (TB) was commensurate with the maintenance performed during the outage. The inspector observed some lapses in good housekeeping practices that resulted in congested areas, making access difficult to certain areas of the plant. However, no areas challenged personnel safety or required immediate attention. Vermont Yankee management focused a significant amount of attention on housekeeping issues during the daily outage meeting by continually stressing the importance of good housekeeping practices and identifying areas requiring improvement. Further, the inspector frequently observed management and supervision performing plant tours and documenting deficiencies. Management also reviewed various personnel accidents to determine whether any changes to their safety policies were necessary. The inspector reviewed VY's housekeeping deficiencies and verified the completion of the corrective actions associated with the issues identified. Based on the overall very good housekeeping conditions within the plant, VY expectations regarding satisfactory housekeeping conditions were well communicated to the outage organization.

2.3.4 Reactor Vessel Inspection and Refueling Activities

The reactor vessel inspections and refueling activities for RO XVI were observed to be well controlled and performed in accordance with approved procedures. The inspector verified procedural prerequisites and sign-offs, observed the satisfactory accomplishment of specific procedural steps, and determined that refueling operators and engineers were knowledgeable of procedural requirements. The inspector noted effective control by the senior reactor operator licensed for refueling activities and effective communications between the refueling operators and Reactor and Computer (R&CE) Department engineers. This coordination was professional and contributed to event-free refueling activities.

The inspector reviewed VY procedures OP 1410, Rev. 17, "Fuel Loading" and OP 1411, Rev. 13, "Core Verification" and determined that the latest revisions were in use. The procedures also reflected the actual performance of the activities on the refueling floor, and were being adhered to by the operators and R&CE Department engineers. During core verification activities, the inspector observed effective control by the R&CE Department engineer of fuel assembly seat checks and fuel assembly identification number checks. The inspector noted a few minor deficiencies in the documentation of refueling core verification activities; however, department supervision were already cognizant of the deficiencies and had initiated appropriate corrections. The records accurately reflected the correct core load. The inspector noted that the R&CE Department assigned a third engineer to the refueling floor during core verification to perform an independent check of the proper loading of fuel assemblies. This was viewed as a positive initiative.

2.3.5 Control of Plant Work Activities

VY requires the control room shift supervisor (SS) to be responsible for the control of systems and equipment during all phases of plant operation. This principle was manifest during RO XVI when the Operations Department identified and prevented the performance of at least two maintenance activities that, if left unchallenged, would have reduced the number of operable core fill systems and would have contributed to a reduction in plant safety. The two cases involved maintenance activities associated with the core spray system and relay testing for loss of normal power supply logic. This excellent attention to detail provided the necessary back-up to planning and scheduling to assure the proper sequencing of maintenance. However, this level of control was diminished in one instance when less than adequate oversight contributed to the improper restoration of a reactor vessel water level instrument following the replacement of an excess flow check valve. This resulted in an ECCS actuation and core flooding with the reactor vessel head removed (Section 2.2.6).

Very good control of outage activities was also reflected by the control of emergent work. This type of reactionary work included maintenance activities to correct: component failures (RMS, service water pump "A," and the "B" EDG); testing or inspection results (motor operated valve testing, local leak rate testing, low pressure feedwater heater maintenance, and the repair of a RHRSW leak); and problem maintenance (RPV head o-ring).

Overall, good planning and scheduling changes were implemented to account for emergent work. The priorities assigned to such items were commensurate with the safety function of the equipment involved. Few maintenance activities or increases in the outage scope could be attributed to poor planning, even though areas for improvement were identified by VY.

3.0 RADIOLOGICAL CONTROLS (71707)

3.1 Inspection Activities

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

3.2 Inspection Findings and Review of Events

3.2.1 As Low As Reasonably Achievable (ALARA) Concern During RHR System Operation

On March 18, the inspector observed three contractors in the immediate vicinity of the lower head of an RHR heat exchanger being operated in the shutdown cooling mode. One of the three contractors was badged for unescorted access and the other two were badged as visitors. The contractor badged for unescorted access was assigned the dual responsibility as the firewatch and as the escort. This required the visitors to stay in the area of the lower RHR heat exchanger head for an additional 30 minutes following the completion of the hot work. The inspector was concerned that the workers were unnecessarily exposed to a higher radiation dose than required to perform the job.

A radiation protection (RP) technician was called to survey the area and to determine whether the radiation exposure to the workers could be reduced. Contact radiation levels on the bottom of the heat exchanger approached 100 mr/hr and general radiation levels were approximately 20-40 mr/hr approximately four feet from the heat exchanger. ALARA postings indicated that the radiation levels were 10-20 mr/hr in this general vicinity. The radiation survey performed on the morning of March 18 indicated 20 mr/hr in the general area next to the heat exchanger.

The inspector concluded that the RP technician properly assessed the condition and provided excellent instructions to the workers on how to reduce their exposures. The inspector discussed this event with RP supervision and was concerned that the contractor visitors received unnecessary exposure to radiation while their escort was assigned to the firewatch in a relatively high radiation area. The RP Supervisor acknowledged the inspector's concern and informed the supervision responsible for the subject personnel of the concerns. The visitors were promptly escorted from the area. In a subsequent discussion, the RP Supervisor indicated that the ALARA practices could have been improved for this particular job; however, he noted that the RP technicians were sensitive to ensuring good ALARA practices. The inspector was informed that it was not the contractor's policy to assign escort duties to a firewatch, and that more effective personnel planning should prevent this in the future. The inspector had no further questions on this matter.

3.2.2 Radiological Housekeeping

During tours of the RB and TB, the inspector observed very good radiological housekeeping practices. Radiological work areas were well posted and provided appropriate information to inform workers of conditions. Frequent RP supervisory tours were observed to immediately correct or enhance radiological postings and work conditions. The assignment of RP technicians to specific areas of the plant contributed to overall good practices and fostered a sense of individual responsibility for their areas. Inspector questions were immediately acknowledged and understood by the technicians. Appropriate responses were initiated, in part, because the technician could relate to maintenance and surveys recently performed in his area. High volume work areas such as the turbine floor, condenser and heater bays, refuel floor, and the first floor of the reactor building met VY management's expectations, as reflected in daily planning meetings, regarding the need to implement good radiological controls and practices. Deficiencies were typically few in number and, when identified by the inspector, were rapidly corrected. The inspector concluded that the radiological conditions during RO XVI were improved from the previous outage and that RP practices were professional and effectively minimized personnel and material contamination.

3.2.3 Radiological Event Response

The RP staff were also observed to be responsive to radiological liquid spills within the plant. During two minor spills, the inspector observed the prompt establishment of radiological boundaries and postings. In addition, the RP staff attempted to identify the cause and evaluate the significance of each event. These initiatives were understood by on-shift technicians and contributed to informing on-coming shift personnel and supervision. Survey activities were promptly initiated and included both the areas and workers in the immediate vicinity. Access to the spill areas was promptly restored following decontamination efforts.

3.2.4 Communication of Contamination Event Report Information

The inspector reviewed the manner in which communications were conducted between the radiation check point (RCP) for the radiologically controlled area (RCA) and the RCP at the drywell access. The inspector noted that there was a lack of communication from the RCA to the drywell RCP regarding the status of personnel contamination event reports and the results of the subsequent surveys. Vermont Yankee requires that radiological surveys be performed following personnel contamination events, in part, to provide an assessment of the contamination levels and the scope of the decontamination effort. However, when contamination events occurred within the drywell, all records were kept at the RCA RCP. Not communicating this information to the drywell RP technicians was reflected in their lack of knowledge of recent contamination events that occurred on jobs prior to their shift that were still on-going. This conclusion was based on the inspector questioning various RP technicians assigned to the drywell access RCP. The technicians could not articulate what previous maintenance activities caused personnel contaminations. However, the technicians were able to inform the inspector of events that happened on their own shift. This observation was acknowledged by various RP technicians as being important to a good pre-job brief and could contribute to preventing subsequent contaminations during similar maintenance activities. The RP Supervisor acknowledged inspector's comments and concerns and indicated that the matter would receive further review.

3.2.5 Respiratory Protection Program Activities

During a facility tour on March 29, the inspector observed respiratory protection equipment inspection and maintenance activities. RP contractor personnel involved in the activities appeared knowledgeable and informed the inspector as to the procedure used to control the activity and the nature of the training they received. Plant procedure AP 0505, Rev. 22, "Respiratory Protection" specifies as a prerequisite that only personnel trained and qualified shall operate the respirator fitting/equipment or perform maintenance on respiratory protection equipment. In addition, Section D, Maintenance of Respiratory Protection Equipment, or the aforementioned procedure requires that replacement of parts or repair will be performed by personnel who have received documented training in the maintenance of the subject equipment. Follow-up review by the inspector of training records identified that VY had not kept the necessary record to satisfy this requirement for a contract senior RP technician. The inspector obtained collaborating information by interviewing involved VY and contract personnel to determine that appropriate training was conducted. The contractor in question was the only non-permanent RP staff member trained and authorized to conduct respirator maintenance. The inspector also questioned the training methods used in performing respiratory protection training for contract technicians, since they appeared to use a far less disciplined approach than the Systems Approach to Training used for in-house technicians.

Following identification of the issues to the RP Supervisor by the inspector, the cognizant RP Assistant was assigned the task to assess the respirator repair training program for contractors. Inspector concerns in this area were referred to an NRC:RI specialist for further review. Refueling outage-related inspection 92-08 conducted April 13-17 documented further NRC review on this matter.

4.0 MAINTENANCE AND SURVEILLANCE (62703, 61726, 92700)

4.1 Maintenance

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

4.2 Maintenance Observations

4.2.1 Enhancements in Maintenance

As part of an effort to improve the performance of certain maintenance during refueling outages, VY acquired special tooling and equipment to reduce personnel radiation exposure, increase personnel safety, and potentially reduce the amount of time necessary to accomplish certain tasks. Two notable improvements were associated with the rebuilding and inspection of CRD mechanisms and the installation and removal of main steam line (MSL) plugs. Both activities have historically resulted in high personnel radiation exposures and the identification of personnel safety concerns.

<u>CRD Rebuilds</u>

With respect to the CRD activity, VY purchased a transportation CRD cask to transfer the radioactive CRD mechanisms from the CRD access hatch to the CRD rebuild room, a distance of approximately 50 feet. Previously, CRD mechanisms were transferred on an open air gurney that exposed workers to higher than necessary radiation levels. Using cask transportation, personnel exposure was reduced, personnel safety was improved, and contamination was better controlled. The CRD would then be removed from the transportation cask and set into the rebuild cask. The rebuild cask uses water and stainless steel shielding for radiation protection and contamination control. Filtered water covers the CRD mechanism and continuously recirculates, flushing the work surfaces of contamination. In addition, improvements were made to the rigging equipment used to transport the CRD mechanisms from their under vessel position to the CRD skid.

The transportation and rebuild casks reduced the radiation exposures and contamination levels for this activity. Specific activities associated with the new equipment contributed to a reduction of approximately 2.4 man-rem. Even with this reduction, the total exposure was comparable to the last RO levels (approximately 12 man-rem) due to increases in the work scope and unplanned material problems. Recommendations to further reduce exposure will be documented by the CRD Task Force Post-Outage Report and considered for implementation.

MSL Plug Installation/Removal

The second activity enhanced was the installation and removal of MSL plugs. The function of the MSL plugs is to isolate the main steam system from the reactor vessel when reactor vessel water level is raised above the level of the pipe openings. The higher-than-normal water level supports reactor refueling. Vermont Yankee acquired special equipment to remotely install and remove MSL plugs from the reactor vessel flange area. This significantly increased personnel safety. Previously, this task was performed by personnel standing on the reactor's steam separator. In addition to the improvement in personnel safety, the use of hand tools directly above the open reactor vessel for this activity is no longer required.

Besides the considerable improvement in personnel safety, the remote installation technique also significantly reduced personnel radiation exposure. There was approximately 0.390 man-rem expended for this activity, or approximately one quarter of the prior exposure for this task. Mock-up training on the new equipment contributed to the effective use of the new equipment and provided assurance that the plugs would be properly installed.

The actions taken by VY to improve the performance of these two activities were positive enhancements to the Maintenance Program, and reflect a strong VY commitment to plant and personnel safety.

4.2.2 "B" EDG Maintenance Associated with the ECCS Tests

Vermont Yankee conducted a total of three ECCS tests to verify the proper operation of the core cooling and electrical systems during a loss of coolant accident coupled with a loss of off-site power. The surveillance associated with this test is documented in Section 4.4.3. The first two tests, conducted April 5 and 7, were unsuccessful due to non-test problems with the "B" EDG. The April 5 test was not successful because the "B" EDG failed to start, due to incomplete resetting of the diesel governor shutdown plunger following the last operation of the diesel on April 3 (when the "B" EDG successfully completed its 8 hour monthly operability test). The failure to start was due to a surface defect on a contact surface used to reset the fuel racks to the no-fuel position. The April 7 test failed because the "B" EDG output breaker did not automatically shut. In this case, the EDG achieved rated speed; but, due to binding on an auto-start relay, a contact failed to reposition to allow the output breaker to shut.

Vermont Yankee preliminarily determined that the root cause for the first failure was the advanced age of the "B" diesel governor which led to excessive wear on a contact surface needed to reset the shutdown plunger. VY identified a popping noise and an uncharacteristic "feel" during the resetting of the governor, and verified that this was directly related to the contact surface and the diesel failing to start. The inspector noted that VY had previously decided that the "B" governor would be replaced prior to this event. This determination, in part, was made during the repair activities and evaluation associated with a problem of the "A" EDG governor to satisfactorily control diesel load back in February 1992. Inspection Report 92-04 documented this condition and the replacement of the "A" EDG governor. However, based on satisfactory performance during surveillance testing and, in part, due to unavailability of parts, VY was reasonably assured that the "B" EDG governor would continue to perform its safety function until its scheduled replacement in May 1992. Both the "A" and "B" governors were original diesel supplied equipment.

Vermont Yankee was assisted by a service representative from the diesel vendor to help troubleshoot the governor and to assist in governor adjustments. The services of the YNSD Quality Assurance Department were also enlisted to provide an independent assessment of work control and quality verification. Based on discussions with the vendor representative, VY decided to replace the governor with one that was recently rebuilt (refere to Section 8.4). On April 7, the rebuilt governor was satisfactorily installed and tested.

The repair efforts for the second failure were associated with the EDG output breaker permissive circuitry. This particular portion of the breaker circuitry is only tested during the performance of the integrated ECCS test, which is performed once per operating cycle in accordance with TS. The inspector noted that both EDGs successfully started and powered their respective safety-related buses in response to the April 23, 1991, loss of offsite power event. This (coupled with satisfactory performance of monthly surveillances) provided reasonable assurance that, up to the time the binding occurred on the relay, the "B" EDG could have fulfilled its intended safety function.

The inspector observed the performance of relay testing and concluded that the maintenance personnel were knowledgeable. The inspector questioned the appropriateness of testing the dc relay with an ac power source and whether a procedure was necessary to ensure correct use of the relay testing equipment and satisfactory documentation of test requirements. The inspector discussed the first concern with the Maintenance Supervisor, who indicated that this had already been identified and additional testing with a dc source would be completed. With respect to the use of a procedure; the inspector subsequently concluded, based on the knowledge demonstrated by the maintenance personnel performing the troubleshooting, that this type of relay testing was within the skill of the craft and did not require a procedure. The inspector verified that the specific testing and troubleshooting requirements, as directed by plant supervision during a maintenance meeting, were satisfactorily documented and resolved in the completed work package.

In regards to both failures, the inspector reviewed the maintenance packages and discussed probable failure modes with VY representatives. Good equipment knowledge was exhibited. This was complemented by a good questioning attitude. In addition, the inspector observed YNSD Quality Assurance Department and VY management and department supervisors questioning the involved VY personnel to gain a better understanding of the failure mechanism and the subsequent repair and retest efforts. This reflected a good safety perspective commensurate with the significance surrounding the failure of two consecutive integrated ECCS tests.

4.3 Surveillance

The inspector performed detailed procedure reviews, witnessed in-progress testing, and reviewed completed surveillance packages. The inspector verified that the tests were performed in accordance with TS, approved procedures, and NRC regulations. The surveillance observed was concluded to be effective with respect to meeting the safety objectives of the testing program.

4.4 Surveillance Observations

4.4.1 "A" Core Spray System Full Flow Test

On March 19, the inspector observed portions of the "A" core spray system full flow test. This test was conducted using procedure OP 4123, Rev. 22, "Core Spray System Surveillance" which is required to be performed during refueling outages by the plant's TS. The test verifies the ability of the "A" core spray system pump to deliver water at a rate of 3000 gpm. This flow rate is verified against a system head which represents the pressure drop equivalent to discharging into the reactor vessel.

The inspector noted that the core spray system was not returned to the normal standby condition per OP 2123, Rev. 21 "Core Spray" as directed by OP 4123. This was due to the normal keep-fill method being unavailable as a result of the plant's condensate system being secured for the outage. An alternate keep fill source utilizing the condensate storage tank was lined up in accordance with procedure RP 2171, Rev. 22, "Condensate Demineralizer System." The inspector noted that RP 2171 needed enhancement to cover system restoration during off-normal plant configurations, such as plant outages. The inspector also noted that the method used to document the position of the valve before and after the conduct of the full flow test was not in accordance with normally accepted practices. The valve position was written on a yellow slip of paper and stuck to the control panel. The issue was discussed with Operations Department management, who acknowledged the inspector's comments and concerns, and indicated OP 2123 would be enhanced to cover system operation during off-normal configurations. Additionally, Operations Department management indicated that other procedures which could be similarly affected would be reviewed and revised as necessary.

The inspector's overall assessment of the "A" core spray system full flow test was that is was adequately performed. Control Room personnel conducting the test understood the procedure and performed it in a safe and competent manner.

4.4.2 (Closed) NCV 92-06-01: Missed TS Surveillance (LER 92-02)

On January 15, with the reactor at 100 percent of rated power, VY identified that the boron concentration check of the standby liquid control (SLC) tank had not been performed within the TS time limits. TS 4.4.C.1 requires that the boron concentration be determined at least once per month and TS 1.0.Y. allows a 25 percent time extension on the monthly surveillance. When the surveillance was performed on January 9, the concentration was within requirements; however, the surveillance interval exceeded the TS time requirements. The test was previously conducted on November 27, 1991.

Vermont Yankee determined that the root cause was a personnel error associated with the scheduling of this particular surveillance. A contributing cause was the lack of an effective review of the surveillance schedule as required by the VY surveillance program. VY procedure AP 4000, Rev. 14, "Surveillance Testing and Control" provides the mechanism for scheduling, independent verification, and documenting completion of required surveillances. To prevent recurrence, VY initiated a procedure revision to AP 4000 to provide additional guidance on the review of the surveillance schedule following the surveillance test coordinator's review. Vermont Yankee also performed various other reviews of the surveillance program to ensure that no similar missed surveillances existed.

Based on: a review of procedure AP 4000; past events regarding missed TS surveillances; and, a discussion with the coordinator responsible for the surveillance program, the inspector concluded that this event was isolated and could not have been prevented by previous corrective actions. VY acknowledged that a lack of attention to detail contributed to this event, and a more thorough review of the surveillance schedule could have identified that the surveillance interval would have been exceeded. This violation will not be subject to enforcement action because the licensee's efforts in identifying and correcting the violation met the criteria specified in Section VII.B.(1) of the NRC's Enforcement Policy (NCV 92-06-01).

4.4.3 ECCS Integrated Automatic Initiation Test: Surveillance Attributes

On April 9, VY successfully completed an ECCS automatic initiation test and verified that the ECCS system would perform as designed. This test is performed once per operating cycle. A preliminary review of test data, plant and system response, and operator observations indicated that all tested components functioned within TS and/or design requirements. The initial test failures and associated repair efforts are documented in Section 4.2.2.

The scope of the ECCS test encompassed: (a) emergency core cooling systems, such as high and low pressure injection, core spray, and reactor core isolation cooling; (b) containment systems, such as the standby gas treatment system; and, (c) emergency electrical power systems, such as the emergency diesel generators, uninterruptible power supplies, and station service batteries. The test was initiated from simulated plant conditions imposed on actual plant sensory equipment and logic subsystems. The signals simulated a simultaneous loss of coolant accident and a loss of voltage condition on the safety-related electrical buses.

Test Coordinator

VY assigned a SS the responsibility of coordinating the testing prerequisites (prereqs) and system line-ups necessary to support this test. This initiative placed a well-qualified individual in a position to understand the status of testing prereqs, the effect of the prereqs on the plant, and the current and expected plant conditions as a result of the test. The test coordinator was also a member of the Outage Planning Group and therefore knowledgeable of outage activities. Effective communications between the test coordinator, the operating crew, and VY management were observed and contributed to the overall understanding of plant conditions. These discussions often focused on the status of the test prerequisites and maintenance that could have potentially affected the test.

Documentation of Test Prerequisites

The inspector independently verified the test prerequisites and compared the results to the documented completion of the prerequisites in the master test procedure. Differences between actual plant conditions and the envisioned plant conditions, as signed off in the master test procedure, were reviewed and discussed with the test coordinator. The inspector observed that the prerequisite sign-off made in the master test procedure did not reflect the current plant condition when reviewed. This matter was discussed with the Operations Supervisor and Operations Superintendent. The inspector verified that, prior to the conduct of the test, necessary prerequisites were completed.

Pre-Test Brief

The inspector observed the pre-test brief conducted by the senior line manager responsible for the overall performance of the test. The brief was well performed and of specific detail to inform the testing personnel and control room operators of expected plant conditions. The brief included the delegation of individual responsibilities and assignments, a review of procedural steps, and was attended by all necessary personnel to assure the safe performance of the test. Appropriate senior management observed the brief and the performance of the surveillance test.

Conclusions

Overall, the inspector found that VY actions taken to prepare and perform the ECCS automatic initiation test were very good, and that the knowledge level of cognizant individuals was good. A high level of attention to detail was sustained by control room operators and testing personnel. A very good pre-test brief was conducted.

5.0 EMERGENCY PREPAREDNESS (71707)

5.1 Emergency Notification System (ENS) Inoperability

On April 9, the inspector informed the control room operators that the resident office ENS phone was not working properly. The control room made a prompt ENS call due to the use of the resident office as an NRC assembly point within the Technical Support Center during plant emergencies. VY procedure AP 0156, Rev. 16, "Notification of Significant Events" describes, in part, that a major loss of emergency communications capability includes the loss of "any red phone." The phone was repaired within a couple of hours.

5.2 Emergency Call-In Response

On April 15, VY improved the ability of the NRC resident inspectors to respond to plant events by providing the inspectors with two radio-communication pagers. The pagers are used in conjunction with the VY emergency call-in procedures to expedite the mobilization of key personnel, during off-normal hours, in the event of a plant emergency.

6.0 SECURITY (71707, 90712, 92700)

6.1 Observations of Physical Security

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

6.2 Fitness-For-Duty For Cause Testing

During this inspection period a number of events involving For Cause testing occurred.

Between the period of March 7-13, four contractors working on-site in a visitor access status were detected by VY security personnel of having alcohol odor on their brea' At the respective times each individual was confirmed positive for alcohol as a result of th. conduct of For Cause testing in accordance with the VY Fitness-For-Duty Policy, VYP:222. Testing conducted included checks for both drugs and alcohol abuse. Because these individuals had either received on-site training in VY Fitness-For-Duty requirements or had previous nuclear industry experience, and therefore should have been aware of the plant's expectations, they were all denied further site access. VY's actions, in these cases, were determined by the NRC to reflect a conservative fitness for duty perspective. The diligent performance of security personnel in identifying potential alcohol abuse was commendable.

On March 29, two contractors were identified by the Security Shift Supervisor (SSS) as having alcohol odor on their breaths. For Cause testing was conducted for alcohol abuse. In both cases the blood alcohol content was below NRC and VY limits. One individual returned to work and the other individual was given time off as a prudent measure. Subsequently, on March 30, the Plant Services Supervisor identified irregularities in the execution of the Fitness-For-Duty Policy. Specifically, For Cause testing was limited to the alcohol portion only. Decisions made by the SSS subsequent to the detection, which resulted in failure to conduct the drug testing portion of the policy, were determined by VY to represent human error. The SSS's actions were contrary to the policy procedure and training provided to members of the security force on the subject, as well as requirements specified in 10 CFR 26.24.

The inspector reviewed the VY investigation report and corrective actions to preclude recurrence. These efforts were of good quality and appropriate to the circumstances. Additionally, the inspector noted that although this event was not a reportable security event, an informational telephone call was made to the cognizant security specialist inspector at the NRC Region I office (NRC:RI) on March 30. This violation will not be subject to the enforcement action because VY's efforts in identifying and correcting the violation meet the criteria in Section VII.B.(1) of the Enforcement Policy (NCV 92-06-02).

On April 10, a VY employee was identified by his immediate supervisor displaying impaired behavior that warranted For Cause testing. The testing, performed shortly after the individual's shift begun, was promptly confirmed to be positive. The individual was escorted offsite and his access was removed. The individual was an unlicensed supervisory person. A review conducted by VY indicated that no work or documentation deficiencies resulted from his condition. In accordance with VY's Fitness-For-Duty Policy, the individual was referred to the Employee Assistance Program and was given a minimum 14-day suspension. VY made a timely 24-hour report to the NRC pursuant to 10 CFR 26.73(a)(2).

6.3 Drywell Access Control

The VY security organization recently implemented a change to their post orders requiring drywell access control security officers to verify that materials intended for use in containment were listed on a restricted chemical list (RCL). If the chemical was on the RCL, the material would require a valid chemical use permit which listed the quantity, authorization signatures, and an expiration date. This permit is authorized by the SS. If the officer determined that the material was not listed on the RCL, he would employ a compensatory measure to verify that the material was necessary for use in containment. The compensatory measure assured that all chemicals were positively controlled and that the proper authorization was presented.

10 CFR 73.55(d)(8) describes the requirements necessary to control the access of materials into containment. The effective implementation of these requirements substantially reduces the vulnerability of critical reactor plant equipment to radiological sabotage. The inspector observed the conduct of security operations at the drywell accesses for personnel and equipment and control rod drive accesses, and concluded that the security officers were knowledgeable of their post orders and effectively implemented their assigned duties. The inspector also verified that VY implemented positive controls to assure only authorized materials were permitted in containment.

6.4 Security Plan Change

On March 27, VY was notified by NRC:RI during a telephone conference that a change (Revision 21) to their Security Plan as submitted under 10 CFR 50.54(p) on February 26 was unacceptable, in that, it constituted a decrease in the plan's effectiveness since a prior VY commitment was removed. The NRC provided written notification to VY on April 9 to submit appropriate changes as necessary to correct the plan. Further pursuit of this matter by VY will require them to submit a request to the NRC in accordance with the provisions of 10 CFR 50.90.

Immediately following the March 27 telephone conference, VY initiated corrective action to ensure that the prior plan's commitment was implemented. Subsequently, the inspector verified the reestablishment of the committed action, that post orders were generated and in-place, and that the officers were knowledgeable of their assigned duties.

The inspector had no further questions on this matter.

6.5 Security Human Factors Consideration

The extensive nature of the planned activities during the RO, and the large number of contract personnel necessary to accomplish the scheduled tasks, resulted in the security organization developing an aggressive outage work schedule. The scheduled work hours for non-supervisory security officers consisted of two 12-hour shifts that worked seven or eight days (with two days off) followed by seven or eight days work (with four days off). This cycle was repeated for the approximate six week outage. The Security Shift Supervisors and alternates worked their normal shift rotations; however, the inspector noted that they were working significant amounts of overtime as well.

Vermont Yankee administrative procedure AP 0894, Rev. 1, "Shift Staffing/Overtime Limits," which lists the security force as part of minimum shift staffing requirements, contains administrative limits on overtime for personnel. These limits were established to address the Commission's Policy Statement on nuclear power plant staff working hours. Included within these limits are instructions that individuals should not be permitted to work more than 72 hours in any seven day period. However, a note in the procedure indicates that the subject limit does not apply during extended periods at cold shutdown, unless the individual is assigned operator duty in the control room. The security contract manager responsible for work schedule development informed the inspector that it was this exclusionary provision of AP 0894 which allowed the establishment of the aggresive outage schedule. This practice was also used during the 1990 RO.

The inspector has observed that VY relied on extensive use of overtime from the prior 1990 RO to the present time. This appears to have resulted from staffing shortages due to attrition, as well as an inability to provide timely staffing replacements to support both Gatehouse 2 modifications and corrective actions for weaknesses identified during the October 8-11, 1991 NRC Operational Safeguards Response Evaluation. The inspector was concerned that the long-term reliance on overtime, in conjunction with an outage schedule that would result in security officers working up to 96 hours in an eight day period, were job-related human factors which could adversely influence the effectiveness of the security organization.

Vermont Yankee security organization representatives acknowledged the inspector's concerns and indicated that they were sensitive to this issue. The VY security organization responded by implementing the following short term actions: (1) providing written instructions to security personnel to inform their supervisors of exhaustion or fatigue that could preclude proper performance of their duties; (2) identifying situations were excess security personnel exist, in an attempt to require security officers to leave work after eight hours; (3) using of security supervision to augment shift coverage; and, (4) ensuring that work requiring security involvement would only occur when staffing was available. There were no morale issues or security performance deficiencies identified by the NRC during this inspection period.

In prior meetings with the NRC, VY indicated that security management was in the process of addressing staffing and overtime issues. Prior to the RO, VY authorized their security contractor to employ temporary personnel to perform watchperson duties and related tasks in an effort to reduce reliance on overtime. However, sufficient progress in providing needed relief was not observed, which suggests that a weakness continues to exist in the oversight of security contractor activities. VY management attention is warranted to address the continuing practice of excessive overtime during long-term shutdown conditions.

7.0 ENGINEERING AND TECHNICAL SUPPORT (71707, 93702, 90712, 90713)

7.1 Reactor Vessel Cladding Indications

On March 26, VY engineering supervision informed the inspector that the inservice inspection (ISI) performed on the internal surfaces of the reactor (BN-1) identified spider-like surface indications in the cladding of the reactor pressure vessel (RPV) head and the vessel flange area. This ISI examination was performed as part of the normal RO inspection cycle. A GE Service Information Letter (SIL No. 539) described the specific characteristics of these indications and recommended actions associated with this issue. Recent nuclear industry experience indicated that the indications were caused by stress relief cracking as a result of service induced stress corrosion and welding of clad to the inner surface of the RPV head and vessel. Preliminary inspections performed by a YNSD Level 3 non-destructive examination inspector indicated that the cladding flaws did not penetrate into the RPV head base metal. In addition, there were no indications that the cladding was separating from the base metal. Extensive testing by VY and YNSD. and documentation which was reviewed by the NRC, concluded that the structural integrity of the RPV was assured and would support reactor operation. Also identified during this inspection was an indication located in the vessel clad near a steam dryer support bracket. This indication was also evaluated to not enter the vessel base material and to not affect the structural integrity of the support bracket.

A review of this issue by an NRC specialist was performed during the period of April 6-10 and documented in Inspection Report 92-07. On April 5, VY submitted a report on this issue to the NRC. On April 17, the NRC accepted the VY evaluation and concluded that the impact of the indications would not pose a safety concern for plant operation up to the next refueling outage. Vermont Yankee will provide a complete plan for future inspections prior to the next RO.

7.2 Voltage Setpoint Drift in Under Voltage Relays

On March 31, during calibration of four under voltage relays on safety-related 4160 Vac buses 3 and 4, VY identified that the setpoints for the relays had drifted low out of the TS band. The relays (27/3Z, 27/3W, 27/4Z, and 27/4W) are designed to drop out on low bus voltage and provide an under voltage signal to the core spray loss of coolant protection circuits to initiate auto start of the EDG (in conjunction with low low reactor vessel water level or high drywell pressure), and actuate a safety bus low voltage annunciator and time delay function.

The safety-related 4160 Vac buses 3 and 4 operate at 3700 $\pm$ 40 Vac, however, the under voltage relays sense a correspondingly lower voltage (transformer ratio of 4200:120) to actuate the low voltage condition. The TS band for actuation is 104.57 - 106.85 Vac. The inspector determined that the voltage drift was low out of the TS band by approximately 0.07, 0.17, 0.67, and 0.97 Vac, respectively. The relays are required to be functionally tested and calibrated once per operating cycle. Following the resetting of the relays on March 31, VY performed a satisfactory functional test as demonstrated by the performance of the integrated ECCS test as documented in Section 4.4.3.

A preliminary review of this condition conducted by VY, concluded that the relays have not experienced similar drift. VY will submit an LER to report this condition. The inspector has no further questions at this time.

7.3 Evaluation of Reactor Mode Switch Concerns

As documented in Section 2.2.1 of this report, an inadvertent PCIS Group I isolation and reactor scram occurred on March 6 due to the failure of reactor mode switch (RMS) contacts to close that should have closed. Immediate actions were taken by VY on March 7 to provide interim compensatory measures to assure that protective design features would be functional to support the activities that could occur at the time the RMS position was changed. At this time, both Plant Operations Review Committee (PORC) and plant management involvement provided the proper safety focus, assuring that no further activity would occur that might possibly result in operation outside of the plant design basis due to concerns about the integrity of the RMS. Reconciling themselves to the fact that reportability requirements would result in conducting a root cause investigation and development of longer term corrective actions, plant management then directed its full attention to the RO. The I&C Department initiated investigating actions. However, based on concerns that the protective design features provided by the RMS may not have been fully operable, the inspector requested VY to reconsider its investigation timetable.

On March 11, a task force consisting of various engineering disciplines and the leadership and full participation of the Technical Services Superintendent was assembled. All RMS position changes were evaluated and, for every contact, verified the function of the contact and performed an assessment of the potential consequences of a failure of the contact to change state when the RMS is repositioned. The task force determined that: (1) the March 6 PCIS Group I isolation and reactor scram were caused by contacts not closing that should have closed; (2) no evidence was found that a contact that opens during the RMS repositioning did or would fail to open; (3) for every indication that existed that a contact was not made up, that it involved a contact that would be expected to be closed upon reaching the intended RMS position; (4) in only one case did a potential failure mode exist where if a contact did not close, a protective feature could be bypassed [involves the repositioning of the RMS from "Shutdown" to "Refuel" and back to "Shutdown," such that, the contact failure to close would not reset the mode switch in Shutdown Scram logic]; and (5) in all other cases the failure of a contact to close does not preclude a protective action from occurring, but could be the cause of an unnecessary challenge to Engineered Safety Features, or it could preclude a desired but not required action. No information was identified that would indicate that VY operated in a condition outside of the plant's design basis due to RMS performance concerns.

Regarding item (5) above, the task force determined that the two RMS changes could have negative ramifications, namely moving the RMS from "Run" to "Startup" or from "Startup" to "Run." The former movement could result in the condition that occurred on March 6. The latter case involves the potential that the 15 percent Average Power Range Monitor Scram or the 40 percent Steam Flow Isolation may not get bypassed as intended. No corrective actions were recommended for this case because RMS failures have not occurred with the switch in the "Run" position, which is believed to be associated with the firm manner in which operators move the switch from "Startup" to "Run."

A number of corrective action recommendations were identified by the VY task force that were intended, to the extent practical, to prevent spurious isolations and scrams and resultant plant transients. These included: (1) revise plant procedures to require that during normal plant shutdown, the RMS is moved from "Run" through "Startup" to "Refuel" rather than stopping in "Startup"; (2) verifying that the "Mode Switch in S/D Scram Bypass" annunciator clears, or that I&C has verified that contact No. 2 is closed to address the "Shutdown" to "Refuel" to "Shutdown" concern specified above; (3) use of the report for additional operator training; and (4) the development of a listing of actions an operator can take to provide added assurance that the RMS contacts represent the position that the switch has been repositioned to.

Past VY and industry reported failures involving the RMS were assessed by the task force. Documents assessed included NRC Information Notice No. 83-42, "Reactor Mode Switch Malfunctions," and VY LER No. 86-08, "Unanticipated Scram During Mode Switch Movement," and a Nuclear Plant Reliability Data System Query Report. All related events and information fully supported the VY assessment that RMS integrity concerns involve the condition of contacts not fully closing when the RMS is repositioned due to their remaining in the interim position that occurs between the respective positions. Vermont Yankee included in LER 92-05 detailed information generated from their task force evaluation. The task force report dated March 16 was reviewed by the PORC on March 18, with supporting recommendations of corrective actions being transmitted to the Plant Manager.

The inspector verified that, prior to the plant startup on April 21, appropriate procedure changes and Standing Orders were in place to provide the necessary guidance and directions to plant operators to preclude concerns identified with operation of the RMS. The evaluation provided by the task force demonstrated strong performance in engineering and technical support capabilities by the operating organization. Short, intermediate and long-term corrective actions were timely and appropriate. The report of the task force and the resulting LER were comprehensive. Proper followup to ensure recommendations are implemented was observed. The efforts of VY to resolve concerns about RMS integrity in a timely and comprehensive manner reflects their strong orientation toward nuclear safety.

8.0 SAFETY ASSESSMENT AND QUALITY VERIFICATION (90712, 90713, 71707, TI 2515/113, 40500)

8.1 Licensee Event Reports

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

- LER 92-03, "Advanced Offgas (AOG) Rupture Disk Temporary Repair Not Within System Design Basis." Inspection Report 92-01 documented the isolation of the AOG system isolation resulting from maintenance activities to replace instrument air filters. The isolation caused a system pressure transient to exceed the burst pressure of the AOG rupture disk. The ruptured disk allowed the release of radiological gases and particulates from the AOG system into the plant ventilation system. Inspection Report 92-04 documented the subsequent actions taken by VY to ensure that the AOG system operated within its design basis and addressed operation of the system with the temporary repair installed.

- LER 92-02, "Missed SLC Tank Concentration Surveillance Due To A Personnel Error When Transferring Due Dates To The 1992 Schedule." This LER was reviewed in Section 4.4.2.

  The inspector discussed VY's preliminary evaluation of YNSD's recommendations with the cognizant engineer and supervisor and determined that VY will implement appropriate corrective actions prior to the start-up of the AOG system following the RO. The actions, in part, included maintenance activities to assure proper instrument loop times and valve leakage rates, the planned installation of new steam jet air ejector nozzles during the next outage, and the receipt of additional rupture disks designed with a small burst tolerance. These actions and controlled operational testing prior to start-up should assure the safe and proper operation of the AOG system following the RO.

- LER 92-04, "High Pressure Coolant Injection System Inoperable Due to Degradation of Station Battery Bus Voltage Caused By Failed Battery Charger Component." Inspection Report 92-04 documented the occurrence of this event and concluded that VY's failure to properly implement portions of station procedures contributed to a delay in the reportability of this event. The issuance of a violation regarding failure to follow procedures was, in part, to promote improvements in VY's performance in the implementation of procedures associated with reportability requirements. The inspection report also documented questions raised by the inspector. VY addressed these questions adequately within the LER text.

- LER 92-05, "Reactor Scram During Shutdown Caused by the Contacts on the Reactor Mode Switch Not Closing As They Should Have." This event was reviewed in Sections 2.2.1 and 7.3. The inspector noted that incorrect information was contained in the reporting requirements section. A VY representative indicated that a revised LER would be submitted to address this issue.

- LER 92-06, "Quarter Scram While Shutdown as a Result of the Wrong Fuses Being Removed For Maintenance." This event was reviewed in Section 2.2.2.

8.2 Supervisory Tours

In addition to the observed RP supervisory tours (documented in Section 3.2.2), the inspector frequently observed VY management and supervision on tour in the plant to assess the status of outage activities, including housekeeping and fire protection activities. The supervisors were observed to witness the performance of surveillance and maintenance activities, and question the workers as to specific portions of the activity, and assess identified problems. The discussions between the supervisors and the tradesmen appeared to promote quality by increasing the worker's understanding of the intent of the activity or of specific procedural steps. The discussions also appeared to encourage communications between tradesmen and supervision. This provided an avenue for management to communicate VY expectations on housekeeping, fire protection, general outage activities, and on-the-job performance. Supervisory tours also contributed to the communication of credible and accurate first-hand information as to the status of outage activities and significant safety concerns. The inspector concluded that management and supervisory tours performed during the outage contributed to the effective performance of maintenance activities, communication of accurate information, and discussion of management expectations to all levels of the organization. This reflected a commendable safety ethic and contributed to the safe and proper operation of the plant during a challenging schedule of activities.

8.3 Guidance Regarding the Loss of Decay Heat Removal

In response to the Plant Manager's (PM) review of the Outage Review Safety Committee report, the Operations Department was assigned to assess the need to provide guidance to operators regarding expected actions during a loss of shutdown cooling (SDC) event. The PM specifically referenced a period in the outage in which only one RHR pump was operable to provide core decay heat removal. Other systems were available to support decay heat removal through the use of core flooding (core spray, condensate storage water transfer, control rod drive flow, fire systems, etc.) and alternate core cooling (fuel pool cooling). In addition, management was sensitive to the fact that maintenance on plant electrical distribution systems had the potential to challenge alternate methods to mitigate a loss of SDC event. The inspector noted that during this period with the RMS in "Shutdown," VY exceeded minimum TS system operability requirements, in part, through the effective planning of outage activities. VY implemented additional initiatives to make operators and supervisors more aware of the challenges imposed on the plant during this phase, as described in Section 8.6 of this inspection report. However, the proposed method of implementation for operator guidance was of concern to the inspector; specifically, the information was not reviewed nor intended to be approved as procedures.

The guidance initiated was intended to further clarify procedural steps and enhance operator understanding of the availability/operability of plant systems necessary to keep the core cooled. This guidance paralleled and amplified the instructions in VY procedures ON 3156, Rev 3, "Loss of Shutdown Cooling" and OE 3101, Rev 7, "RPV Control Procedure," and appeared to be based on industry safe shutdown recommendations. The guidance listed the specific steps of applicable plant procedures and provided supplemental information regarding system status and alternate modes of system line-ups. The Operation Department's intention appeared to be to provide diverse shutdown strategies to assure the success of a particular task. The inspector noted that the procedures intended to mitigate a loss of SDC were abnormal operating procedures and did not provide detailed instructions concerning operator response to a loss of RPV level or SDC. ON 3156 and OE 3101 were written to apply to all phases of plant operation, as opposed to during shutdown conditions.

In response to the issue, the Operations Supervisor (OS) restated that: (1) the procedures in place were adequate to provide sufficient instructions to operators for the loss of decay heat removal or reactor vessel level during RO XVI; and, (2) that the guidance was only intended to further clarify and enhance operator understanding. In an effort to resolve the issue, the OS attached a memorandum to provide specific instructions to the operators dictating that the guidance was not a procedure and that it should not be followed as such. This in effect would prevent the use of this guidance to change plant conditions or expected operator actions during off-normal events without the formal reviews and approvals required of plant procedures. The inspector had no safety concerns with this action, but was concerned about the reluctance to implement procedural changes to address specific equipment and/or system line-ups.

8.4 Review of VY Quality Control Documents

During the maintenance efforts necessary to ensure the proper operation of the "B" EDG during the integrated ECCS tests (described in Sections 4.2.2 and 4.4.3), the inspector reviewed the quality control documents associated with the vendor surveillance and receipt inspection of the "B" EDG governor. Vermont Yankee delegates the responsibility of vendor surveillances to YNSD. Final receipt and quality verification of the component is the responsibility of VY. The EDG governor was inspected at the vendor's facility in accordance with YNSD procedure OQA-XVIII-4, Rev 16, "Vendor Surveillances." Commercial dedication was performed by VY under the provisions of VY procedure VYP:330, and receipt inspection was performed in accordance with AP 6015, Rev. 7, "Receipt Inspection of Safety Class or Safety Related Materials."

The inspector identified a need for improvement involving the documentation of inspection performed by YNSD at the vendor's facility, in that the documentation for the diesel governor did not fully describe the "results" of the surveillance on the Vendor Surveillance Summary form, as required by YNSD procedure OQA-XVIII-4, step 6. The completed surveillance form, however, did document the "findings" and "release conditions" as required. Hence, reliance on the field inspector to accurately communicate deficiencies provides the basis for acceptance of equipment at VY.

The inspector discussed this observation with the manager of vendor surveillances at YNSD and noted that the intent of the Vendor Surveillance Summary form was to expedite the receipt of a component necessary for plant operation. Therefore, the form is intended to provide the fundamental information necessary for a satisfactory VY receipt inspection. The formal Vendor Surveillance Report that follows within 30 days of the inspection, completes the documentation of detailed and amplifying information regarding critical attributes and items noted in the surveillances plan. A YNSD Senior Engineer responsible for vendor surveillances acknowledged that the incorporation of a pro-active statement regarding the "results" of the vendor surveillance, coupled with the findings and acceptability determination, would provide a more accurate and explicit documentation tool to be used during VY receipt inspections.

The inspector concluded that the receipt inspection of the "B" EDG governor was performed in accordance with approved procedures. In addition, based on the vendor surveillance performed, a good level of assurance existed that the governor would meet its intended safety function.

8.5 Review of RHR Service Water Leak Activities

On March 13, VY held a meeting to discuss a service water piping leak in the vicinity of a weld for a pressure instrument line. The leak was in ASME code piping near RHRSW-89B motor operated valve located on the service water discharge side of the "B" RHR system heat exchanger. At this time, only the "B" RHR system was operable. The "A" RHR system was inoperable due to preventive maintenance being performed.

The Outage Manager initiated this meeting to discuss the repair efforts necessary to restore the "B" RHR system to a non-degraded status and the effects of this condition on the overall operation of the plant. Even though the plant was in the "Shutdown" mode of operation and did not require shutdown cooling in accordance with the plant TS, management recognized the importance of decay heat removal (DHR) capability. Emphasis was placed on the recognition and availability of alternate core cooling modes and the prevention of maintenance activities that: (1) would have the potential of initiating a plant event requiring enhanced DHR processes, and (2) would prevent operation of alternate core cooling systems. This included the status of off-site power supplies and the status of plant electrical maintenance. Responsibilities for the determination of heat-up rates, temporary and permanent fixes, and root cause were established. Management expectations regarding contingency plans to establish alternate cooling modes and the need to continually trend fuel pool and reactor temperatures were developed.

The actions taken by VY to establish communications between engineering, operations, and maintenance personnel contributed to a well coordinated effort to understand the cause of the RHR SW-89B valve failure, its effect on plant operations, and those activities necessary to support the corrective maintenance repair activity.

8.6 Temporary Instruction (TI) 2515/113: Reliable Decay Heat Removal (DHR) During Outages

Background

Temporary Instruction (TI) 2515/113 was used to review activities during RO XVI which had the potential for contributing to a loss of capability to remove decay heat from the reactor. The TI also required the review of VY processes regarding the reduction or loss of reliable electrical power supplies to support the systems necessary for DHR. Recent industry experience indicates that increased emphasis to manage risk during shutdown operations, and to carefully plan and control maintenance and surveillance, is warranted to assure safe operation during shutdown conditions.

The inspector discussed VY planning and scheduling activities with plant management and outage coordinators. Discussions were held with plant engineers and operations personnel regarding the specific portions of outage planning related to DHR. The inspector reviewed schedules and procedures to understand management expectations regarding the availability of DHR and electrical power systems. The following procedures were reviewed:

OP 2123, Rev 21, "Core Spray System"
OP 2124, Rev 27, "Residual Heat Removal System"
OP 2126, Rev 21, "Diesel Generators"
OP 2140, Rev 19, "345 KV Electrical System"
OP 2141, Rev 11, "115 KV Switchyard"
OP 2142, Rev 13, "4 KV Electrical System"
OP 2144, Rev 18, "120/240 Vac Vital Bus"
OP 2145, Rev 16, "Normal and Emergency 125 Vac Operation"
OP 2146, Rev 14, "Operation of Station and Alternate Shutdown System 125-Volt Battery Chargers"

Vermont Yankee considers the requirements set forth in the TSs as the principle written guidance for outage safety; however, VY plans and controls outage activities to meet or exceed these requirements. This philosophy is supplemented by the incorporation of industry guidance into planning and scheduling. This is intended to enhance margins of safety and improve the sequencing of activities around the availability of plant systems necessary to ensure DHR, containment integrity, and the availability of diverse and redundant electrical power supplies. In order to realize this philosophy and ensure its effective implementation, VY undertook a series of initiatives to better prepare for RO XVI, as described below.

Procedures

Regarding DHR procedures, VY reviewed and implemented many recommendations from industry, NRC Information Notices, and Generic Letters. These procedure changes enhanced precautions associated with reactor coolant inventory control, described industry experience regarding potential reactor vessel drain paths, and reflected management expectations. Administrative limits were also established to provide additional guidance regarding the availability of residual heat removal service water pumps to prevent a loss of DHR capability.

Based on the maintenance and tests performed during RO XVI, special test procedures were not necessary to control decay heat removal system activities. However, when required, VY will initiate special test procedures. This action forces management review of the procedures and appropriate oversight leading to the implementation of the specific activity.

Outage Planning

Management involvement in industry (NUMARC) workshops and committees provided recommendations to improve the planning and control processes associated with an outage. In order to provide first hand knowledge of industry guidance, VY outage planners and coordinators attended these workshops and seminars to improve their perspective of the safety significance of this issue prior to RO XVI. The use of inter-department correspondence documented the reviews performed and the knowledge gained involving safe shutdown issues. Resulting recommendations to reduce shutdown risk were formally reviewed by the Plant Manager, and commitment items were assigned and dispositioned. Recommendations that were not applicable or already incorporated into controls for plant activities were justified as such.

Systems necessary for the continual removal of decay heat, above the minimal TS requirements, were maintained. This was analogous to the industry guidance of one more than the required number of systems. Support systems availability, such as ac and dc electrical power supplies and service water, were also planned with this philosophy in mind. Comprehensive component-level maintenance schedules for these support systems complemented the schedules associated with DHR systems. These efforts ensured the minimum requirements set forth by the plant TSs, as well as an appropriate level of confidence that personnel errors and equipment malfunctions would not impact the safe operation of the facility.

Outage schedules included: (a) *logic-line drawings* sequencing activities in relation to their effect on other outage activities, containment integrity, and reactor plant systems; and, (b) three-day bar charts that, in part, itemized outage activities, length of activity, and the organization responsible for accomplishment. The schedules were continuously updated prior to and during the outage. Further, a daily determination of the "operability" of plant systems was distributed to plant personnel and control room operators for use as a quick assessment tool to determine the status of systems necessary for the function of DHR, for supplying electrical power, and for providing containment integrity.

The inspector reviewed the outage schedule and discussed various portions with VY representatives to determine whether periods of increased vulnerability coincided with minimal availability of electrical power sources. Again VY implemented industry guidance regarding maintenance activities during periods of increased vulnerability. These periods were times during reduced reactor vessel inventory or when a minimal number of systems were operable for reactor fill or heat removal. Specific maintenance activities that were planned around periods of increased vulnerability included CRD mechanism bolt change-out, replacement of CRD mechanisms, activities effecting the status of fuel pool gates, and the identification of work associated with potential reactor vessel drain paths.

Electrical Power Availability

The inspector also reviewed VY controls and practices to ensure the continued supply and distribution of electrical power to DHR systems. Refueling Outage XVI was performed, as planned, with one onsite and one offsite electrical power supply available at all times. During "Refuel" mode VY maintained two EDGs operable when the TS required only one. Further, during all phases of the outage, VY maintained either both EDGs or one EDG and the Vernon 4.16 KV tie operable. The Vernon tie is a reliable, non-safety class electrical tie directly from a local hydro electrical plant to the vital buses. Maintenance in the 345 KV and 115 KV switchyards was conducted in a manner so as to not impact the availability of redundant power sources to the plant.

Vermont Yankee implemented their temporary modification process to control the use of non-standard electrical line-ups. Plant operating procedures provided guidance on the operation of electrical systems using installed cross-ties. These procedures were based on original plant design and analysis. The use of temporary diesel generators to power in-plant loads was also controlled by the temporary modifications program.

In the area of battery power supplies, VY maintained TS operability requirements and supplemented the dc electrical network by providing power that was "available" vice "operable." Other enhancements included: use of in-plant cross-ties, the scheduling of system outages to correspond to electrical bus maintenance; and, the implementation of temporary modifications to supply power. For example, when battery maintenance or testing is performed, VY will consider the respective dc loads as inoperable. However, VY will provide dc power via cross-ties or the use of the bus-specific battery charger, to maintain the dc loads "available." Similarly, when dc power is not available to the EDGs for field flashing or protection relays from its normal source, VY will consider the diesel inoperable until power from an independent safety-related station battery can be supplied.

Training

Operator training and the use of alarm response and emergency procedures provides guidance to operators during off-normal situations, such as a loss of offsite power or the loss of power to specific load centers powering DHR systems. VY places high reliance on training to provide operators appropriate knowledge to promptly implement correct action during events that are not covered by specific plant procedures. For RO XVI, VY did not conduct formal operator training on the different plant configurations that were experienced during the outage. However, previous training had focused on specific outage evolutions and safety-related industry experiences. Management discussions with the control room operators prior to the outage focused appropriate attention to the challenges that specific outages phases would present. In addition, management expectations regarding the control of work effecting DHR and electrical power systems were specifically discussed. This adhoc training provided insight as to the importance of maintaining DHR in-light of recent industry experiences.

Management Initiatives

VY employed an Outage Safety Review Committee to review the outage schedule, identify potential safety issues, evaluate the operability of systems with respect to TSs, and provide recommendations intended to increase the margin of safety during various phases of the outage. The committee consisted of persons licensed as senior reactor operators, risk assessment and system engineers, and a former maintenance supervisor. The recommendations generated by the team were reviewed by the PORC and dispositioned by the Plant Manager. Specific recommendations were adopted that changed the outage schedule to: (1) improve the availability of ac and dc electrical buses; (2) reduce the vulnerability of the plant to a loss of reactor vessel inventory due to maintenance activities; and, (3) ensure the operability of emergency core cooling systems.

An Outage Planning Group was also employed that included supervisory control room operators and shift supervisors acting as system experts responsible for the overall performance of maintenance. This management initiative required the licensed operators to independently assess the effect of maintenance activities on safety-related systems. Specific attention was given to DHR and electrical systems to support operability determinations.

9.0 MANAGEMENT MEETINGS (30702)

9.1 Preliminary Inspection Findings

At periodic intervals during this inspection, meetings were held with senior plant management to discuss preliminary inspection findings. A summary of findings for the report period was also discussed at the conclusion of the inspection on May 11. No proprietary information was identified as being included in the report.

9.2 Region Based Inspection Findings

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

Date	Subject	Rpt. No.	Inspector
4/6-10/92	ISI and Water Chemistry	92-07	P. Patnaik
4/13-17/92	RO Radiological Control	92-08	D. Chawaga