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REGION I

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Facility:	Vermont Yankee Nuclear Power Station
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EXECUTIVE SUMMARY

Vermont Yankee Nuclear Power Station
NRC Inspection Report 50-271/96-11

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a six week period of resident inspection; in addition, it includes the results of announced inspections by two regional engineering inspectors and a regional radiation protection inspector.

Operations

The VY staff's troubleshooting of the No. 1 turbine stop valve testing failure on December 24 was promptly and appropriately executed, with good supervisory oversight of field activities. The Operations staff's lack of recognition of the turbine stop valve's potential reactor protection system trip logic input inoperability was viewed as an operator training deficiency.

The cold weather protection program at VY establishes measures that are generally adequate to protect safety-related systems against extreme cold weather. However, the licensee's 1995 program evaluation, as well as historic and recent weather-related events, indicate that vulnerabilities still exist.

VY staff identification and prompt corrective action to proceduralize operator actions to address degraded grid voltage conditions reflected a thorough examination of design basis documentation and associated NRC commitments. The prompt response by the Operations staff to resolve this potential plant transient vulnerability was noteworthy. Accordingly, the failure to implement a 1986 commitment to the NRC for addressing degraded grid voltage conditions via manual operator action was not cited.

Maintenance

The licensee had appropriately evaluated installation of the two spare breakers for the core spray pump motor during routine preventive maintenance. The licensee's decision to re-install the original breaker and correct the minor problems with the spare breakers vice install them, per the maintenance plan, demonstrated a conservative approach to maintenance.

Surveillance testing was performed well. The inspector observed good communications, procedure adherence, and utilization of training opportunities. Minor material and procedural deficiencies did not detract from the conduct of testing, but indicated areas for improvement with respect to operator attention to detail.

VY staff's identification of the body-to-bonnet steam leak on steamline drain valve MS-V77 was good and their efforts to repair the valve via Furmanite injection were well conceived and controlled, but not achievable. Pending repair alternatives appeared to be receiving proper planning and safety focus.

The I&C and Maintenance departments appropriately track and trend the preventive and corrective maintenance work items. Periodic trend and Work Order summary reports use by station managers was demonstrated to the inspector and found to be effectively implemented in the prioritization of work activities and the allocation of available resources. A sampling review of the outstanding corrective and preventive maintenance Work Orders identified no discrepancies with respect to the proper dispositioning of safety system operability impact.

Engineering

The inspector noted excellent preparation of presentation packages, comprehensiveness of the presentations, the expertise and ownership attitude displayed by the principles, and the ability of the PORC members to participate in discussion of the plant operational issues and collectively arrive at an action decision.

Modifications reviewed were of good quality and were supported by sound bases and thorough reviews to conclude that the changes did not involve an unreviewed safety question. Extensive engineering involvement was noted between offsite Design and onsite Engineering departments. Test procedures developed for performing post-modification tests were appropriate for controlling equipment and were completed satisfactorily. Design requirements were appropriately satisfied.

The absence of differential pressure protection in the EDG room conflicts with the FSAR design and is a condition for which a 10 CFR 50.59 evaluation was not performed. In that this condition was licensee identified and promptly addressed, this issue is being treated as a non-cited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy. The VY staff has not developed the long-term resolution of this design discrepancy. Accordingly, the long-term resolution of this issue will be reviewed in a subsequent inspection period and an Inspector Follow-up Item has been assigned. **(IFI 50-271/96-11-01)** Refer to Section E2.2 of this report.

VY engineering had conservatively limited maximum torus water temperature to a value that allowed continued safe operation of the plant until concerns regarding the discrepancy between the Technical Specification limits were resolved.

Prompt action was taken to verify the safety of continued operations when a HELB calculation, using conservative assumptions, determined that the EQ temperature limit for electrical equipment in the reactor building might be exceeded. The inspector will review the licensee's final corrective action for this condition in a future inspection. **(IFI 50-271/96-11-02)** Refer to Section E2.2 of this report.

The licensee implemented an acceptable procedure for temporary repair of valve MS-V77 until a permanent repair or replacement of the valve can be effected. The temporary modification controlling this repair effort was consistent with regulatory requirements for leak sealing using non-metallic compounds for temporary cessation of leakage.

The BMO curriculum training was effectively conducted. Valuable information was presented to enable engineers to appropriately assess plant deficiencies and perform operability evaluations per established procedures.

The engineering work needed to support identified plant issues was appropriately included on the major project work list and engineering personnel knew the established schedule. The inspectors noted that VY had initiated actions to enhance the major project work list acceptance criteria for inclusion and prioritization of work.

VY continued to develop performance engineers for monitoring system performance.

The QA reviews were very valuable for identifying important engineering issues requiring management review. The inspectors determined that the audit scope was thorough and findings were well-supported. The inspectors noted VY's efforts to assess other YNSD associated plant issues via the establishment of the Joint Quality Audit Group.

VY's commitment to develop and document safety-significant design bases for critical plant systems continued to progress. The methodologies established for documenting and validating design bases for plant systems were found to be consistent with NRC guidance.

Plant Support

The VY staff did not violate the Technical Specifications for periodic sampling and analysis of EDG fuel oil as stated in LER No. 96-29, dated January 3, 1997. Notwithstanding, their corrective actions to address the peripheral VY staff performance concerns were observed to be appropriate and effective.

Vermont Yankee continued to maintain an overall effective program for occupational radiation protection including external exposure controls, internal exposure controls, control of radioactive materials and contamination, surveys and monitoring, and the program to maintain personnel radiation exposures as low as is reasonably achievable (ALARA).

Audits and appraisals by the licensee's staff continued to improve the quality of the program. Facility tours indicated that good controls were established for radioactive materials and contamination, high radiation areas, and very high radiation areas with the exception of one problem identified with an improper high radiation area barrier. External and internal exposure controls effectively limited the total dose assignment to personnel. Decontamination and clean-up after the refueling outage was very good. However, the radiation protection staff was challenged by the steam leak in the torus room that spread contaminated water to various elevations and work locations. The inspectors noted a slow response by the licensee's staff to isolate and contain the radioactive liquid. The ALARA program contributed to a reduction in personnel radiation exposures during the last refueling and maintenance outage.

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DETAILS

Summary of Plant Status

Vermont Yankee (VY) operated at 100 percent reactor power throughout this inspection period with the exception of power reductions to conduct planned rod pattern exchanges and surveillance testing.

I. Operations

O1 Conduct of Operations[1]

O1.1 Review of Turbine Stop Valve Testing Failure

a. Inspection Scope (71707)

Using the guidance of inspection procedure 71707, the inspectors reviewed the licensee's response to the failure of the No. 1 turbine stop valve to stroke via a test signal during surveillance testing on December 24.

b. Observations and Findings

During the mid-shift on December 24, the operators commenced periodic surveillance testing of the turbine stop and control valves per Operating Procedure (OP)-4160, "Turbine Generator Surveillance." The No. 1 turbine stop valve did not respond to the test signal, but the remaining three stop valves functioned as designed. The inspector determined that the control room operators promptly notified responsible Operations and Maintenance management and a troubleshooting action plan was formulated to start early on the day shift. Prior to the commencement of troubleshooting, the inspectors discussed the troubleshooting plan with the I&C manager and then observed its implementation from the control room. Troubleshooting identified that the mechanical linkage to the hydraulic test pilot became disengaged, thereby preventing hydraulic fluid pressure from being bled-off to permit valve closure by the test signal. ALARA considerations prevented on-line repair, but the valve was manually stroke tested satisfactorily (following a Plant Operations Review Committee approved procedure change). The licensee plans to repair the linkage during the next convenient unit outage.

The inspectors observed that in the intervening hours between discovery of the failure of the No. 1 turbine stop valve to stroke and the successful manual stroking of the valve following discovery of the test linkage problem, that the control room operators did not consider the No. 1 turbine stop valve (TSV) inoperable and therefore its associated reactor protection system (RPS) logic input inoperable. Each TSV provides input (ten percent valve closure) to an RPS trip logic. The closure of any combination of three TSVs will result in a reactor scram signal. Technical Specification Table 3.1.1 identifies the minimum number of operating instrument channels per trip system to be two, but exempts turbine stop valve closure output relays from being manually tripped should they be inoperable because this action may result in the tripping of the trip system. The inspectors determined that even if

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

the control room operators had considered the inoperability of the No. 1 TSV RPS input function, no action would have been required, except administratively tracking the TS entry condition.

The inspectors discussed the above observation with Operations department management and learned that similar TSV problems had occurred in the past. The Operations staff (licensed operators) had not previously considered the inability to stroke the TSVs to result in questioning the operability of the TSVs' RPS trip logic input. However, they agreed that their past practice and their actions on December 24 were not consistent with an appropriate application of the TS action statement for an inoperable TSV closure function. As a result, action was initiated to train all licensed operators on the correct implementation of this TS requirement and the future dispositioning of similar RPS logic input potential or confirmed inoperability determinations.

c. Conclusion

The VY staff's troubleshooting of the No. 1 turbine stop valve testing failure on December 24 was promptly and appropriately executed, with good supervisory oversight of field activities. The Operations staff's lack of recognition of the turbine stop valve's potential reactor protection system trip logic input inoperability was viewed as an operator training deficiency.

O2 Operational Status of Facilities and Equipment

O2.1 Cold Weather Preparations

a. Inspection Scope (71714)

The inspector reviewed the licensee's program to protect safety-related systems against extreme cold weather.

b. Observations and Findings

The governing procedure for VY's cold weather protection program is OP-2196, "Preparations for Cold Weather Operations." The procedure is to be initiated when average ambient temperatures are less than 35°F, weather forecasts indicate the onset of cold weather, or during the week of October 15. Required preparations for the operations, maintenance, and instrument and controls (I&C) departments are specified in attachment 1 to OP-2196, "Cold Weather Initiation Operations Checklist." Following completion of this checklist (nominally during the first two weeks in November), the operations department performs a verification of cold weather preparations using attachment 3 to OP-2196, "Operations Cold Weather Protection Verification Checklist." Subsequent verification that cold weather protection measures are in place and operating properly is performed by operators during normal watchstanding tours. Normal operations are restored when weather conditions dictate, normally during the first week in April.

The inspector reviewed OP-2196, revision 9, effective September 16, 1996. The inspector noted that the attachment 1 checklist appeared to cover all susceptible areas of the plant. The inspector noted that heat trace operability checks were to be performed per a plant

drawing. From discussion with the electrical maintenance foreman, the inspector determined that the types of checks that were performed were appropriate, however, the results of these checks were not documented. Given the large number of circuits involved (eight freeze protection panels, each supplying up to 24 heat traces), the inspector considered that a procedure would be appropriate for performing heat trace operability checks. This would ensure all circuits were tested and would document individual results. In addition, the inspector noted that heat trace thermostat setpoints were not verified as a part of OP-2196. Finally, some items include a note that the established condition may be changed as required (for example, operation of the reactor recirculation units); other items, such as freeze protection for the fire hydrants, may be disturbed by normal use. However, it was not clear what process ensures that the required cold weather condition will be reestablished.

Previous Licensee Audit Results

During review of OP-2196, the inspector noted that several of the changes that were made by revision 9 were based on a January 1995 Quality Assurance audit, VY-95-02. The audit report noted that several weather-related equipment failures during the winter of 1993-1994 suggested the need for a review of the plant's cold weather policies. The report made the recommendation (SSCA 1106) to, "initiate a comprehensive evaluation to determine basis and adequacy of severe weather controls to preclude similar events." VY concurred with this recommendation and assigned an evaluation completion date of September 1995. The evaluation was tracked as item 1 under VY's response to the audit, Manager of Operations Implementing Directive 9502 (MOOID9502-01).

The inspector reviewed VY's evaluation of severe weather controls, MOOID9502-01, dated October 4, 1995. The evaluation was performed by the operations department using two plant personnel and one contract individual. The evaluation report included a walkdown of significant areas of the plant, comparison of the VY cold weather operations procedure with those of two other nuclear power stations, and a review of industry events. The report concluded with 25 recommended corrective actions in areas such as system performance evaluations, procedures, equipment modifications, maintenance, and trending. These recommendations were assigned as items 8 through 32 under MOOID9502.

The inspector reviewed the status of implementing the MOOID9502-1 recommended corrective actions. As of this inspection, 15 items were closed and four additional items had been generated as offshoots of original items. Most of the remaining open items were significantly overdue (greater than six months). The inspector reviewed the completed corrective actions and considered that, in general, they adequately addressed the stated concerns. An exception, however, was the response to item 16, "Develop a surveillance, Verification of Preparation for Cold Weather Operations. This would include periodic monitoring of ambient temperatures in certain locations susceptible to freezing, verifying the effectiveness of heat trace circuits, the inspection of insulation integrity, etc." The response was to revise OP-2196 with a new checklist (attachment 3, "Operations Cold Weather Protection Verification Checklist") to incorporate a yearly plant walkdown. The inspector considered that the response was weak, in that it did not address the concern for periodic monitoring.

Additionally, the inspector focused on the licensee's position in response to an item relating to the Final Safety Analysis Report (FSAR). Item 29 stated, "Determine if a FSAR change and/or a 50.59 review is required due to the house heating boiler (HHB) steam pressure currently set at 38-40 psig. The FSAR states a pressure of 50 psig." In an extensive response, the licensee demonstrated that, "even with the present steam pressure settings, the boilers should be capable of providing sufficient heat input to the HVAC system to meet all FSAR section 5.3 and 10.12 temperature requirements." The response went on to recommend resetting steam pressure to a higher value (although still less than 50 psig) to provide additional margin for operation during changeover between the two boilers. The response concluded that, "no further safety or engineering evaluations should be necessary as the FSAR and HELB studies have already evaluated the system pressure at 50 psig." While the response appeared to adequately support operation of the HHBs at reduced pressure, the decision to operate in a manner other than described in the FSAR must be supported by a 50.59 evaluation and the FSAR must be changed. VY management acknowledged this oversight and initiated action to conduct a 50.59 evaluation and revised the FSAR. These failures to perform a safety evaluation per 10 CFR 50.59 and revise the FSAR constitute violations of minor significance and are being treated as non-cited violations, consistent with Section IV of the NRC Enforcement Policy.

Review of Cold Weather Events from 1995-1996 Season

The inspector reviewed the index of the licensee's corrective action program event reports (ERs) for the period November 1995 through March 1996 and, based on the summary event descriptions, reviewed several ERs that appeared to be related to cold weather conditions.

ER 95-675 - On November 25, 1995, investigation of an abnormal reading on demineralized water storage tank level indication determined that the level transmitter sensing line was frozen and that the strip heater was de-energized. The cause was determined to be that the wire to the strip heater had failed open due to inadequate sizing. Long term corrective action was: (1) to determine if undersized original installation wiring for outdoor panel/cabinet heaters was a generic concern, and if so, to take corrective action, and (2) to develop a list of outdoor panels/cabinets which utilize strip/block heaters for freeze protection, and revise OP-2196 to energize and perform an inspection/functional test.

The inspector noted that OP-2196, revision 9, does not include the recommended change. In addition, the evaluation referred to MOOID9502-1, stating that, "Even if the corrective actions assigned for [MOOID9502-1] were completed, this event could not have been prevented." However, the inspector considered that the periodic surveillance recommended by MOOID9502-16 would have improved the chances of detecting the malfunction before it became an operational concern.

ER 96-008 - On January 8, 1996, the house heating fuel oil jelled causing clogging of the oil pump suction filters. This resulted in HHB trips and a loss of reactor building ventilation. The cause was determined to be a receipt of a delivery of fuel oil rather than blended fuel, which then jelled due to low temperature in the outside storage tank. Corrective action was to ensure that the Purchasing department ordered blended fuel during cold weather periods.

As a result of this event, the heat trace for the piping from the tank into the plant was checked and found to be defective; specifically, the thermostat was wired such that the heat trace energized during warm conditions and de-energized during cold conditions. This was not listed as a contributing cause in this ER, but was the subject of a separate ER, 96-009. This ER remains open pending determination of whether use of heat trace on the system is appropriate. The inspector considered that failure to identify the malfunctioning heat trace as part of cold weather preparations per OI-2196 indicated that an operational verification of heat trace thermostats would be appropriate.

ER 96-091 - On February 6, 1996, freezing conditions in the chemical treatment shed caused the supply line to the "B" auxiliary chemical treatment pump to freeze. The ER notes that the room is poorly insulated with no winterization around doors and window louvers. No long term corrective action was proposed, and the ER remains open.

Cold Weather Walkdown

On January 14, the inspector performed a partial walkdown of plant buildings to assess the extent and effectiveness of cold weather preparations. Outside temperature was approximately 20°F. The following observations were made:

- Intake structure, circulating water (CW) pump room - roof ventilator IRV-3 (which also opens the room east side louvers) and room heater IEUH-2 were both cycling. Thermostats for the two running units are close to "C" CW pump, and the motor was blowing warm air on them; IRV-3 eventually lowered the room temperature to the point that it overcame the effect of the warm air blowing on the IEUH-2 thermostat, and the heater would turn on. The thermostat for the other roof ventilator, IRV-4, is on the opposite end of the room, about twice as far from A-CW pump (which was secured at the time). Temperature indication on the IRV-4 thermostat was pegged low. The inspector informed operators of this observation and was informed that it was a known deficiency that was being addressed by a minor modification. While permanent corrective action was appropriate, the inspector considered that the thermostats could have been adjusted on the basis of which CW pumps were operating, such that the proper room temperature would be maintained in the interim.

- Intake structure, service water pump room - no problems noted. The room heater cycling and neither roof ventilator was cycling.

- John Deere diesel room - no problems noted. The room was warm, the heater was running, and there was a thermometer in room for operators to monitor temperature. The fuel oil storage tank is buried next to building, and thus is not susceptible to cold weather affects; piping runs underground directly into the building, to a transfer pump and a small day tank.

- Outside radioactive waste storage tanks, freeze protection panel (FPP) 1A - Indication on the FPPs consists of a power available light and 24 indicating lights for individual heat trace circuits. From discussions with the electrical maintenance foreman, each panel is controlled by a single thermostat which energizes and de-energizes all active heat traces. A dark circuit light on an energized (thermostat

on) panel indicates a defective heat trace, a circuit that is not in use, or a burned out bulb. For FPP-1A, the inspector noted that the indicating lights for circuits 23 and 24 were out. The inspector subsequently determined that these were spare circuits, however, this was not indicated on the panel.

- Temporary diesel fuel oil storage tank heated enclosure - Heater #4 disconnect closed light was out.

- Condensate storage tank - FPP-1B had lights out for circuits 4, 11, and 22.

- Condition of outside piping insulation was overall very good. The inspector did note that some canned insulation on piping associated with the HHB fuel oil storage tank was opened.

The above observations were discussed with the responsible department managers and station management.

Recent Cold Weather Events

During the inspection period, three events occurred that were related to cold weather conditions.

- Multiple trips of reactor building ventilation due to a single freezestat (January 1)

- Service water intake bay bubbler failed due to loss of electrical power (January 8)

- Ice accumulation caused intake structure water level to drop by 5.5 feet (January 8)

These events illustrated that opportunities for improvement remain, with respect to the existing measures taken by the VY staff to prepare for, and operate during, periods of extreme cold weather.

c. Conclusions

The cold weather protection program at VY establishes measures that are generally adequate to protect safety-related systems against extreme cold weather. However, the licensee's 1995 program evaluation, as well as historic and recent weather-related events, indicate that some vulnerabilities still exist.

O7 Quality Assurance in Operations

O7.1 Degraded Grid Voltage Operator Actions

a. Inspection Scope (93702, 71707)

On January 2, 1997, the Engineering department's Design Basis Documentation group initiated an Event Report (ER No. 97-0007) which identified that VY had not implemented a commitment to the NRC. The commitment involved the proceduralizing of operator manual actions to address degraded grid voltage conditions which could potentially affect station

4160 volt emergency buses and their associated safety systems. The inspector reviewed the circumstances involving this licensee identified non-conforming condition and the corrective actions taken by the VY staff.

b. Observations and Findings

The inspector determined that the NRC staff had issued a Safety Evaluation, dated March 31, 1986, which concluded that VY's response to an NRC Generic Letter, dated June 3, 1977, was acceptable. By VY letter to the NRC dated January 22, 1986, the licensee agreed to implement plant operating procedures to cover operator action under degraded grid conditions. Operating Procedure (OP)-3140, Alarm Response - Alarm Annunciator Response Sheet: "Low grid voltage without an accident signal," was proposed to be revised and implemented prior to restart from the 1985/1986 outage.

VY staff examination of the current revision (4) of the Alarm Response Sheet for annunciator 8-J-9, "4Kv Distribution: Safety Bus Voltage Low," issued March 29, 1995, identified that none of the prescribed operator actions were incorporated into the operator actions section. The inspector determined from the Operations staff that all of the Alarm Response Sheets were upgraded a few years ago to an improved format. Efforts to locate earlier revisions to the annunciator 8-J-9 Alarm Response Sheet were unsuccessful and therefore, inconclusive as to the precise cause for the operator actions not being incorporated in revision 4. The licensee did identify that comparable operator actions (similar to those actions proposed and accepted in the March 31, 1986 NRC safety evaluation) were proceduralized in the current revision of Off-Normal Procedure (ON)-3155, Loss of Auto Transformer, issued November 16, 1995. However, the inspector verified with the Operations staff that this procedure would not necessarily be referred to for an emergency bus degraded voltage condition.

The inspector observed, on January 2, 1997, that the Operations staff took prompt action to develop a revision to the annunciator 8-J-9 Alarm Response Sheet. The intent procedure change was reviewed and approved during an emergency meeting of the Plant Operations Review Committee and approved for use by the Plant Manager prior to the end of the work day. The inspector reviewed the revision and found that it acceptably implemented the NRC staff's 1986 safety evaluation findings. This licensee identified and corrected non-compliance (TS 6.8, Procedure adequately established) is treated as a Non-Cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy.

c. Conclusion

VY staff identification and prompt corrective action to proceduralize operator actions to address degraded grid voltage conditions reflected a thorough examination of design basis documentation and associated NRC commitments. The prompt response by the Operations staff to resolve this potential plant transient vulnerability was noteworthy. Accordingly, the failure to implement a 1986 commitment to the NRC for addressing degraded grid voltage conditions via manual operator action was not cited.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Maintenance Observations

a. Inspection Scope (62707)

The inspectors observed portions of plant maintenance activities to verify that the correct parts and tools were utilized, the applicable industry code and Technical Specification requirements were satisfied, adequate measures were in place to ensure personnel safety and prevent damage to plant structures, systems, and components, and to ensure that equipment operability was verified upon completion of post-maintenance testing.

b. Observations and Findings

The inspectors observed portions of the following maintenance activity:

- Breaker replacement for "B" core spray (CS) pump motor, observed on January 8

The original breaker was to be replaced with a spare for the purpose of performing routine preventive maintenance on the original. However, the spare breaker was found not to fit properly in the cubicle; specifically, there was inadequate clearance between the interference blocks (seismic restraints) on the breaker and in the cubicle. A second spare breaker was available, but the decision was made not to install it based on minor scalloping on one of the main line contactors. The original breaker was reinstalled and verified to operate satisfactorily.

c. Conclusions

The licensee had appropriately evaluated installation of the two spare breakers for the core spray pump motor during routine preventive maintenance. The licensee's decision to re-install the original breaker and correct the minor problems with the spare breakers vice install them, per the maintenance plan, demonstrated a conservative approach to maintenance.

M1.2 Surveillance Observations

a. Inspection Scope (61726)

The inspector observed portions of surveillance tests using inspection procedure 61726 to verify proper calibration of test instrumentation, use of approved procedures, performance of work by qualified personnel, conformance to limiting conditions for operation (LCOs), and correct post-test system restoration.

b. Observations and Findings

The inspectors observed portions of the following surveillance tests:

- CS system full flow quarterly surveillance test, observed on January 8.

The inspector observed good communications and good self- and second checking of procedural steps. Test results were satisfactory.

During the test, the inspector noted a material problem with the "B" CS pump torus suction isolation valve, CS-V14-7B. Specifically, one of the two bolts that secure the two packing gland follower studs to the valve did not have a nut installed. Based on appearance, the inspector considered that this condition had existed for a long period of time, and did not represent an immediate operability concern. The inspector reported the condition to the shift supervisor. A priority one work request to install the nut was initiated, and an event report was generated to investigate the deficiency.

- Reactor core isolation cooling (RCIC) system pump operability and full flow surveillance test, observed on January 16.

This test was performed at a reduced interval (six weeks) due to pump vibration in the alert range during a previous test; the normal test frequency is quarterly. Operations personnel also utilized the test for initial auxiliary operator qualification training. Test results were satisfactory, with the exception of one pump vibration reading (test point O-4) which was in the alert range. Also, the inspector noted minor steam leakage from a RCIC supply steam trap. The exact location of the leak could not be determined due to flashing that is installed around the trap for personnel protection. A work order was submitted to investigate this problem. During review of the surveillance test procedure (OP-4121), the inspector noted several minor procedural inconsistencies which were promptly addressed by the licensee.

c. Conclusions

Surveillance testing was performed well. The inspector observed good communications, procedure adherence, and utilization of training opportunities. Minor material and procedural deficiencies did not detract from the conduct of testing, but indicated areas for improvement with respect to operator attention to detail.

M1.3 Review of Body-to-Bonnet Leak on MS-V77

a. Background

During startup of the unit from the Fall 1996 refuel outage, the Operations staff identified what appeared to be a small steam leak from the packing gland of MS-V77 (main steam line drain outside containment isolation valve) and initiated a work order (WO) repair request (#26815) on November 4, 1996. Because tightening the packing on a motor-operated valve potentially affects the disc-to-seat thrust, the packing gland was not immediately adjusted and a WO package was planned which required coordination between several departments. The MS-V77 packing was tightened on November 13 and it was then determined that the steam leakage was from a body-to-bonnet seal leak. A re-scoped repair

effort was immediately initiated and Event Report 96-1106 was written on November 14 identifying the inoperability of MS-V77 (degraded containment boundary). Corrective action was taken to tag shut MS-V74 (MS-V77's paired inside containment isolation valve) per TS 3.7.D.2 and to verify and log daily, per TS 4.7.D.2, the closed position of MS-V74.

Following preparation of temporary modification (TM) No. 96-047 (see section E2.3 of this report), a Furmanite injection repair to the body-to-bonnet seal area was attempted on November 22 to stop the leak. This repair was not completely successful and following additional engineering reviews a second injection was made on December 5 and 6, 1996. The second injection temporarily stopped the steam leak, but by December 11, the leak worsened. At that point, the VY staff abandoned any further efforts to repair the valve by Furmanite injection methods and initiated planning for mechanical repair of the valve (i.e. valve internals replacement, seal welding the body to the bonnet, or valve replacement). The VY staff was still developing these repair contingencies at the conclusion of this inspection period.

b. Inspection Scope (62707, 37551)

The inspectors monitored VY's Furmanite repair activities to verify proper application and control of this temporary sealant repair method. The inspectors reviewed VY's efforts to control the steam leakage and to quantify the steam leakage in an attempt to support a Basis for Maintaining Operability (BMO) of MS-V77. In addition, the inspectors monitored the VY staff's deliberations of available repair options.

c. Observations and Findings

The inspectors observed and reviewed various aspects of the Plant Operations Review Committee's review and approval of TM No. 96-047, and of the TM's implementation. The inspectors noted appropriate sensitivity to industry lessons learned with temporary sealant repairs, good administrative controls, and proper supervisory oversight of individual sealant applications. The inspector noted good use of thermography to assist in monitoring the leak, leakage progression, and in verifying the source of steam from the downstream side of the valve vice valve seat leakage.

The inspectors observed the licensee's efforts to quantify the steam leakage to support a BMO for declaring MS-V77 operable for use in the event of a main steamline isolation valve (MSIV) closure (main steamline drains are used to equalize pressure around the MSIV prior to opening). The licensee experienced considerable difficulty in constructing an enclosure around MS-V77 and then collecting and quantifying the amount of leakage. The inspectors challenged the performance engineering staff's plan and their ability to equate steam leakage to a comparable Appendix J leakage acceptance criteria, and therefore be able to conclude with a reasonable degree of confidence that MS-V77 satisfied its containment isolation function (operable). However, the performance engineers abandoned this effort because the ability to properly contain and subsequently quantify the amount of steam leakage was not achievable.

The inspectors monitored the plant staff's efforts to contain/control the steam leaking from MS-V77 and observed that VY was not initially aggressive in their efforts. As a consequence, the steam condensed unabated in the affected sector of the torus room and

resulted in a broad area of potential contamination as the condensate dripped from the overhead onto the torus shell and surrounding piping and cable trays, down onto the torus room floor, and without any channeling to a floor drain in an adjacent sector. This condition also resulted in numerous nuisance fire detection system alarms due to the steam adversely affecting the area smoke detectors. With additional management focus, this condition was remedied as the enclosure was improved and portable ventilation fans and duct work were configured to better contain and condense the leaking steam. (Also see Section R3.3 of this report)

The inspectors monitored planning meetings to examine various repair alternatives and noted broad departmental participation, open and frank discussion of options and their respective advantages and disadvantages, and a cautious and safety-conscious approach to resolution of this operational problem. As of the close of the inspection period, four possible repair contingencies were still being developed and no definitive date of repair established. The inspector periodically monitored the MS-V77 enclosure and steam condensing apparatus in the torus room and observed no appreciable degradation or increase in steam leakage.

c. Conclusion

VY staff's identification of the body-to-bonnet steam leak on steamline drain valve MS-V77 was good and their efforts to repair the valve via Furmanite injection were well conceived and controlled, but not achievable. The staff was not aggressive in implementing actions to initially contain the steam leakage. Pending repair alternatives appeared to be receiving proper planning and safety focus.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Maintenance Work Order Backlog and Trending Review

a. Inspection Scope (62707)

Using guidance of inspection procedure 62707 the inspector conducted a broad review of the preventive and corrective maintenance work order (WO) backlog. The inspector also reviewed the Maintenance and Instrumentation and Controls (I&C) departments' trending of all open WOs.

b. Observations and Findings

The inspector determined through review of open WO reports and discussions with the Maintenance and I&C managers that the current total WO backlog is approximately 2,600 items. The WO backlog consists of pending corrective maintenance (outage and non-outage) and preventive maintenance (scheduled and overdue) on both safety and non-safety related components, system, and structures (inclusive of general maintenance such as building repairs and housekeeping items).

The inspector reviewed the 261 corrective maintenance items outstanding for the I&C department (as of January 3, 1997) and determined that these items are reviewed weekly by the I&C manager and supervisors for prioritization and resource allocation. The I&C

manager stated that this number of outstanding corrective maintenance items was higher than normal (performance goal less than 100) due to two vacant technician positions (soon to be filled) and a build-up of non-outage work items during the recent refuel outage. The balance of the I&C department's approximate 1,000 WO backlog included routine preventive maintenance (139), calibration (425), engineering design or support WOs (155), and routine surveillance testing (7). The inspector verified that the I&C WOs were being appropriately tracked and trended by VY management. Inspector examination of outstanding I&C department corrective maintenance identified no operability concerns that weren't already being monitored via the TS limiting condition for operation (LCO) tracking system (i.e., advanced offgas system recombiner train "A" hydrogen monitors Nos. 2921A and 2922A, and containment hydrogen/oxygen monitor system 2).

Review of the current Maintenance department WO backlog identified approximately 1600 total items (down from approximately 2800 WOs pre-outage), which included preventive maintenance (1145), corrective maintenance (172), and general maintenance (286) work items. Discussion with the Maintenance manager and review of weekly trend reports identified that the total non-outage WOs (preventive and corrective maintenance) and outage WOs varies predictably with the unit operating cycle and resources are managed accordingly. A review of several months of weekly trend reports demonstrated that the total number of WOs was significantly impacted by outage-related preventive maintenance WOs. The inspector notes that the Maintenance department staff pays particular attention to the preventive maintenance backlog greater that 25 percent overdue via a weekly status report. In addition, a recently implemented preventive maintenance discrepancy report provides explanations for the overdue preventive maintenance WOs and completion dates. As of December 16, there were eleven overdue preventive maintenance items, but the inspector noted no adverse operability impact on safety-related systems.

The inspector reviewed a current open WO summary report and examined eleven safety-related systems to determine if the outstanding WOs had any adverse impact on system operability. As mentioned above, those WOs listed which did affect system operability were properly dispositioned. The inspector identified no discrepancies and confirmed that the majority of the open WOs were assigned to preventive maintenance activities.

c. <u>Conclusion</u>

The I&C and Maintenance departments appropriately track and trend the preventive and corrective maintenance work items. Periodic trend and Work Order summary reports use by station managers was demonstrated to the inspector and found to be effectively implemented in the prioritization of work activities and the allocation of available resources. A sampling review of the outstanding corrective and preventive maintenance Work Orders identified no discrepancies with respect to the proper dispositioning of safety system operability impact.

M8 Miscellaneous Maintenance Issues

M8.1 (Closed) IFI 96-09-05: Primary Containment Nitrogen Purge System Isolation Valve Leakage

As reported in inspection report 96-09, on September 24, a combined leak rate test of containment isolation valves for the primary containment nitrogen purge system indicated excessive leakage from one or more of the valves. An inspector followup item was initiated to review the licensee's root cause determination and corrective action. In inspection report 96-10, the inspector reviewed Licensee Event Report (LER) 96-025, "Inadequate testing leads to mis-adjustment of isolation valve mechanical stop and failure to meet Technical Specification (TS) leak rate limits for containment purge isolation valve." The licensee's investigation revealed that SB-16-19-10 was the source of the leakage. This valve is a 18-inch air-operated butterfly valve. The cause of the leakage was determined to be that the actuator mechanical stop was not properly adjusted and prevented the valve from achieving full closure. The mis-adjustment had not been identified in earlier leak tests due to the test method and the valve orientation. This item remained open due to inconsistencies between the LER and the event evaluation that had been developed through the licensee's corrective action program (Event Report No. 96-825).

During this inspection period, the licensee submitted supplement 1 to LER 96-025. The supplement revised the event root cause and long term corrective actions to be consistent with the determinations reached in ER No. 96-825. Along with inadequate test methodology, a second root cause was determined to be lack of configuration control in the valve orientation. Long term corrective actions included inspection, and reorientation if necessary, of similar valves to ensure that containment pressure assists in valve seating. This action is expected to be completed prior to startup from the 1998 refueling outage.

The inspector considered that the root cause determination adequately bounded the problem, and that the corrective actions were appropriate. Deferral of the orientation verification is acceptable, given that all other valves were leak tested in the accident direction during the 1996 refueling outage. This verified that, regardless of any other consideration (valve stop adjustment or valve orientation), valve leakages were within 10 CFR 50 Appendix J limits. In addition, the inspector verified that the vendor's manual states that these valves will provide tight shutoff at less than 150 psi with flow in either direction.

The inspector concluded that the leakage of SB-16-19-10 was of minor safety significance. LER 96-025 stated that, "Historical leak rates were reviewed for the outboard primary containment isolation valves in the lines common to SB-16-19-10. This review showed that since 1978 the applicable pathway leak rates were within TS limitations and accident assumption values." Therefore, primary containment remained intact throughout the period in question. This licensee identified and corrected violation of Technical Specifications individual valve local leak rate criteria is being treated as a Non-Cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy.

M8.2 Licensee Event Report Reviews

a. Inspection Scope (92700)

Using the guidance of Inspection Procedure 92700, the inspectors reviewed the Licensee Event Reports (LERs) discussed below to verify the VY staff had implemented the corrective actions, as stated in the LERs, and to determine whether their response was appropriate and met regulatory requirements.

b. Observations, Findings, and Conclusions

- LER 96-019, Half scram and group III containment isolation caused by loose reactor protection system breaker termination, dated September 25, 1996.

  This unplanned engineered safety feature (ESF) actuation occurred on August 27, 1996, and was discussed in inspection report 50-271/96-08. The event occurred due to a trip of the "A" reactor protection system (RPS) trip logic supply breaker, 5ACB1A. This breaker supplies power to the "A" RPS trip channels and portions of the primary containment isolation system (PCIS). The licensee determined that the root cause of the event was improper installation of the circuit breaker during initial plant construction, in that non-standard termination techniques were used on the cable that connects the breaker to the bus. One of these connections had come loose, which resulted in excessive heating and caused the breaker to trip. As corrective action, the breaker was inspected and tested, and was reinstalled using new termination hardware. The "B" RPS trip logic supply breaker was also inspected. Its terminations were also found to be non-standard, but tight. Following its return to service, the "A" RPS breaker was examined using thermography, and found to be operating hotter than expected. As a result, the "A" breaker was replaced and the "B" breaker was re-terminated during the 1996 refueling outage.

  The inspector assessed that LER 96-019 met regulatory requirements. The event had no safety significant consequences, and the root cause would not prevent the RPS and PCIS from fulfilling their safety functions. The licensee's description of the cause of failure was thorough and the corrective actions were appropriate.

- LER 96-025, Inadequate testing leads to mis-adjustment of isolation valve mechanical stop and failure to meet Technical Specification leak rate limits for containment purge isolation valve, dated October 24, 1996, and LER 96-025, Supplement 1, dated January 3, 1997.

  This event and the licensee's corrective actions are discussed in section M8.1 of this report. The inspector assessed that these reports were of good quality and satisfied regulatory requirements. LER 96-025 and Supplement 1 are closed.

- LER 96-006, Supplement 1, Potential inoperable residual heat removal service water valves due to the bolts holding the valve operators being insufficiently tight, dated November 27, 1996.

The original LER was reviewed in inspection report 50-271/96-03. Supplement 1 to the LER was submitted to document the root cause of the event, which was determined to be a manufacturing-fabrication deficiency in that the vendor did not adequately torque or use a thread locking material or device. LER 96-006, Supplement 1 is closed.

III. Engineering

E1 Conduct of Engineering

E1.1 Engineering Interface with Other Plant Functions

a. Inspection Scope (71707)

The inspector attended a regular meeting of the Plant Operations Review Committee (PORC) to observe and assess the quality of engineering interface with other plant functions toward resolution of significant plant issues.

b. Observations and Findings

Membership of the PORC represented plant management, site and headquarters engineering, operations, maintenance, I&C, radiation protection, and licensing. Principles for each issue provided documentation describing the issue, and responded to questions by the PORC members.

A wide range of significant plant issues were discussed at the meeting. These included:

Potential Degradation of Secondary Containment
Appendix J Program Discrepancies
Reactor Cavity Shield Blocks
4KV Breaker Degradation
Torus Temperature Limitation Evaluation

Following the discussion for each item, the PORC assessed the issue and collectively approved further action by the principle or directed further investigation by the principle.

c. Conclusions

The inspector noted excellent preparation of presentation packages, comprehensiveness of the presentations, the expertise and ownership attitude displayed by the principles, and the ability of the PORC members to participate in discussion of the plant operational issues and collectively arrive at an action decision.

E2 Engineering Support of Facilities and Equipment

E2.1 Engineering Design Change Reviews

a. Inspection Scope (37550)

The inspectors performed reviews of selected safety significant engineering design change requests (EDCRs) previously implemented at VY. These reviews were performed to verify that the modifications were technically sound and that the impact of the changes were thoroughly evaluated. Additionally, these reviews focused on the quality and effectiveness of installation and test procedures for satisfying design requirements and for verifying that the changes did not create an unreviewed safety question. Portions of the following EDCRs were reviewed:

- 96-412 SSPV and SDV Vent and Drain Pilot Valve Replacements
- 96-401 Appendix R ASD Fuses
- 95-409 Service Water System Water Hammer
- 95-402 Replacement of Feedwater Heater C-5-1A/B

b. Observations and Findings

- 96-412 Scram Solenoid Pilot Valves (SSPV) and Scram Discharge Valve (SDV) Vent and Drain Valve Replacements

This plant modification replaced the SSPVs and SDVs to improve control rod scram response time. The inspectors found that the replacement of such components had many complex attributes including generic industry applicability, environmental qualification concerns, and unique sub-component susceptibilities as documented in the following NRC inspection reports.

93-09	94-13	94-18	95-10
95-25	96-03	96-05	96-06

The inspectors discussed the history, design development, testing specifications, and qualifications of the new SSPVs and associated components installed with the cognizant instrumentation and controls engineer. The inspectors found that VY had originally identified the hardware and manufacturing flaws associated with the exhaust diaphragm sub-component of the valves in 1989, and was instrumental in the development and implementation of the replacement valves installed during the fall 1996 outage. The original design valves were determined by VY to be the root cause for many past challenges for satisfying plant technical specifications, and for unscheduled plant shutdowns at VY because scram time requirements could not be met.

The inspectors reviewed the safety assessment, 10 CFR 50.59 evaluation, and post-modification installation and testing procedures developed for this engineering change. Based on this review, the inspectors found that the engineering involvement between

Design Engineering and onsite Instrumentation and Controls Engineering was extensive and well-coordinated. The inspectors reviewed completed anticipated transient without scram (ATWS) and scram time test results of the new design valves and found that excellent improvements had been made that easily satisfied technical requirements of 0.358 seconds for average scram time. The resolution by VY of this design issue and other emergent technical issues associated with the valve replacements was found to be noteworthy and reflected positively on the engineering organization with respect to enhancing plant operations and safety. The inspectors found that design information and requirements presented in the technical specifications, FSAR, and General Electric and Institute of Electrical and 'Electronics Engineers (IEEE) standards were appropriately incorporated and satisfied.

- 96-401 10 CFR Part 50, Appendix R Alternate Shutdown (ASD) Fuses

The purpose of this modification was to install new fuse blocks, fuses, and cables within equipment required for 10 CFR Part 50, Appendix R ASD of the plant. The inspectors reviewed the adequacy of the post-modification installation test procedure written to support closure of the modification.

The inspectors found that the post-modification test procedure was of adequate quality and the testing was satisfactorily completed. Proper independent verifications were found to have been made by the Project Engineer or approved alternate individual(s). The inspectors noted that tagouts were properly controlled within the testing procedure and Quality Control was appropriately involved and utilized.

- 95-409 Service Water System (SWS) Water Hammer

As a result of concerns reflected in NRC Inspection Reports 92-81, 93-04, and 94-18 for possible water hammer in the SWS piping system under several accident scenarios, the licensee performed evaluations VYC-855, VYC-1220, VYC-1220, VYC-1330, and VYC1487 of water hammer potential in the SWS. The licensee found that the SWS could be susceptible to water hammer during a Station Blackout (SBO) event.

The inspector found that the licensee dynamic evaluation of water motion in the SWS demonstrated that column separation (the activator of a water hammer event) might occur from the existence of low pressure in the pipe in the elevated sections of the SWS, particularly downstream of throttled valves during the accident scenarios.

The inspector reviewed the corrective action taken by the licensee to mitigate the probability of water hammer in the SWS system through installation of vacuum breakers at elevated sections of pipe. These provide atmospheric openings at the computed low pressure pipe regions to prevent the pipe internal pressure from falling to the level that initiates water hammer.

The inspector reviewed the detailed EDCR description, safety evaluation, and 10 CFR 50.59 evaluation, and found them to have appropriately described the design criteria, description of change, and organizational responsibilities in implementing the change.

The inspector reviewed the Installation and Test Procedure for EDCR 95-409, Water Hammer Vacuum Breakers, SPN-9522-700. This procedure defined the installation locations, safety classifications, seismic classifications, materials and welding processes, fabrication standards, hydrostatic and leak test procedures, and as-built dimension documentation requirements. The inspector reviewed the completed sign-off sheets for installation and testing.

The inspector found EDCR 95-409 to have been technically justified for ameliorating the possibility of water hammer during design basis accident scenarios. The design change was planned well and implemented in accordance with ASME Code and 10 CFR requirements.

- 95-402 Replacement of Feedwater Heaters C-5-1A/B

The inspectors reviewed the completed replacement of the E-5-1A/B low pressure feedwater heaters performed to achieve greater resistance to carbon steel erosion/corrosion, improved thermal-hydraulic performance, and improved water level stability.

The inspector found VY inspections and evaluations had indicated that erosion/corrosion problems with the existing (carbon steel) heaters made their replacement necessary. The replacement heaters were fabricated of a combination of carbon steel, stainless steel, and chrome-alloy materials to resist corrosion. The design of the replacement heaters was improved to allow increased drain flow capacity and improved water level stability, thereby improving performance and reducing longer range erosion/corrosion. The replacement heater designs were compatible with the revised operating characteristics of the low pressure turbine replaced in 1995.

The inspector reviewed the scope of the replacement feedwater heater design change and quality assurance program. The design changes were indicated in the VY FSAR, Revision 13, VY Technical Specifications, and were in accordance with appropriate sections of ASME, ASTM, ANSI, ASNT, and Nuclear Regulatory Guidance. The inspector reviewed the Engineering Safety Assessment and 10 CFR 50.59 Evaluation and found that the new designs satisfied ASME Code and 10 CFR requirements.

The inspector reviewed the performance test program for the replacement feedwater heaters. In addition to structural and thermal efficiency testing, the test program included operational monitoring to adjust the water level stability performance. To date, the operating performance tests indicate improved performance over the original design. Further improvements are expected from water level adjustments obviated from observed performance behavior. The feedwater heater replacement was performed efficiently through comprehensive technological redesign, good planning, and effective implementation in accordance with Code and Regulatory requirements during refueling outage RO9. The foresight displayed in replacement of the feedwater heaters shows good technological evaluation of design details to provide for an improved regenerative heater system.

c. <u>Conclusions</u>

Modifications reviewed were of good quality and were supported by sound bases and thorough reviews to conclude that the changes did not involve an unreviewed safety question. Extensive engineering involvement was noted between Design and onsite I&C

Engineering Departments within VY. Test procedures developed for performing post-modification tests were appropriate for controlling equipment and were completed satisfactorily. Design requirements were appropriately satisfied.

E2.2 Basis for Maintaining Operability Reviews

BMO 96-08 Emergency Diesel Generator (EDG) Tornado Protection

a. Inspection Scope (37550)

The inspectors reviewed corrective action status for a licensee finding that a tornado pressure relief design for diesel generator and fuel oil day tank rooms had not been implemented.

b. Observations and Findings

As a direct result of NRC Information Notice 96-06, a review of plant tornado pressure relief panels by VY performance and design engineers found that the tornado pressure relief panels described in Ebasco as-built Drawing G191624 had not been installed. Since the Ebasco drawing had been revised to add the tornado pressure relief panels in 1970, the EDGs were not protected from the effect of design tornado pressure differential across the masonry walls since initial plant operation. During this time, without tornado pressure relief panels, plant operation was outside the design basis requirements of FSAR Sections 2.3.6.3 and 12.2.1 to sustain pressure differential loads from tornados.

The licensee took immediate action to insure EDG operability in the absence of the room pressure relief panels. They issued BMO 96-08, "The Effect of Design Basis Tornado Pressure Load on the Diesel/Day Tank Room Enclosures." The BMO described the deficiency and equipment involved, identified the potential adverse effect on safety caused by the condition, considered factors that compensate for the adverse safety impact, gave recommendations and time estimates for correcting the condition, and gave a conclusion and supporting basis that there was reasonable assurance that the plant may continue to safely operate.

The licensee took immediate compensatory measures which included blocking open the EDG room doors and posting fire and security watches. In BMO 96-08, Revision 1, further detailed actions were outlined for the restoration of operability of the diesel and day tank enclosures during the period November 1, 1996 through March 31, 1997 with the doors closed. Design engineering is targeting April 1, 1997 for a permanent solution. The inspector verified that the recommended compensatory measures identified in BMO 96-08, Revision 1 were properly implemented by the plant staff.

c. Conclusions

The absence of differential pressure protection in the EDG room conflicts with the FSAR design and is a condition for which a 10 CFR 50.59 evaluation was not performed. In that this condition was licensee identified and promptly addressed, this issue is being treated as a Non-Cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy. At the close of this inspection period the VY staff had not developed the long-term resolution

of this design discrepancy. Accordingly, the pending long-term corrective actions will be reviewed in a subsequent inspection period and an Inspector Follow-up Item has been assigned. **(IFI 50-271/96-11-01)**

BMO 96-05 Increase in Maximum Torus Temperature Post-LOCA

a. Inspection Scope (37550)

The inspector reviewed the VY Safety Evaluation for Increase in Maximum Torus Temperature Post-LOCA from 166°F to 176°F to allow increased initial torus temperature from 90°F to 100°F.

b. Observations and Findings

In 1995, VY Design Engineering determined that Technical Specification Amendment 88 had not appropriately incorporated the increase in initial torus temperature from 90°F to 100°F into appropriate FSAR sections. An adverse condition (PAC 96-02) cited by VY engineering indicated that modified assumptions, related to system heat transfer assumptions by General Electric (GE) could increase the maximum torus temperature to 176°F with an initial torus temperature of 90°F, contrary to the original prediction of 166°F.

The safety evaluation assessed the effect of change in maximum torus temperature post-LOCA on all accident and transient conditions. LOCA included all postulated breaks within the primary containment as affecting condensation capability, torus loads, and temperature and pressure response. ECCS performance was evaluated including EQ of ECCS and their support systems, ECCS Pump NPSH, Torus instrument accuracy, and ECCS pipe stress and pipe support loads.

The inspector reviewed the VY safety evaluation and Standing Order #19 to administratively limit the torus water temperature under normal or test operation to 87°F, pending resolution of the temperature discrepancies between the Technical Specification 3.7.A.1.a and FSAR Figure 14.6-7. Furthermore, Operations will be notified if the limit is exceeded, and an orderly shut down process will be initiated. The inspectors found the safety evaluation and corrective action to be consistent with regulatory requirements.

c. Conclusions

VY engineering had conservatively limited maximum torus water temperature to a value that allowed continued safe operation of the plant until concerns regarding the discrepancy between the Technical Specification limits were resolved.

BMO 96-18 Main Steam Tunnel Blowout Panels

a. Inspection Scope (37550)

The inspector reviewed the licensee's response to a preliminary engineering determination that the existing arrangement of blowout panels in the main steam tunnel may cause the environmental qualification temperature limit for equipment in the reactor building to be exceeded during some high energy line break scenarios.

b. Observations and Findings

VY is in the process of updating their high energy line break (HELB) analyses for environmental qualification (EQ), using an improved analytical model. In reviewing the existing analyses for HELBs in the main steam tunnel, a problem was noted with the pressure values at which the blowout panels in the main steam tunnel were assumed to actuate.

The main steam tunnel runs through the reactor building, from the drywell to the turbine building. There are four blowout panel configurations associated with the main steam tunnel: (1) Two relatively small panels (about 25 ft^2 total) that blow out to the turbine building at 0.25 psid, designed to accommodate a HPCI/RCIC steam line break; (2) a large area plate (about 85 ft^2) that blows out to the turbine building at 2 psid; (3) a small panel (about 9 ft^2) that blows out to the reactor building at 2 psid; and 4) the door between the main steam tunnel and the reactor building, that blows out at 2 psid. The last three configurations were designed to accommodate a main steam line break. The tolerances in the relief setpoint vary significantly between the four configurations due to different designs. The HPCI/RCIC blowout panels are held shut by counterweights, and therefore relieve at a known pressure with a tight tolerance (within 0.075 psi of the set pressure). The panel between the main steam tunnel and the reactor building relieves by shearing fabricated aluminum studs, which gives it a wider tolerance (1.8 to 2.1 psid). The remaining two configurations relieve by shearing the rivets that hold them in place, which gives them a still wider tolerance (although assumed not to relieve at less than 2.0 psid).

The problem that was noted was that the original calculation for a HPCI steam line break did not take into account tolerances in the pressures at which the blowout panels would relieve. The revised calculation determined that the peak pressure that would occur in the main steam tunnel would be 1.9 psig. Using the design relief values, this would only cause the HPCI/RCIC blowout panels to relieve, releasing steam only to the turbine building; however, taking the set pressure tolerance into account, this peak pressure could also cause the 9 ft^2 panel to the reactor building to relieve. Were this to occur with the reactor building initially at its maximum allowable temperature (100°F), the new calculation showed that the resultant peak temperature would exceed the EQ temperature limit for some electrical equipment that was required to operate following a HPCI steam line break.

As a result, an administrative limit of 80°F was established for average reactor building temperature, with the actual temperature to be determined and recorded daily. This was instituted by an operations department standing order (SO #23, dated December 27, 1996). A BMO was developed to verify that continued operation was acceptable. BMO 96-18, "Main Steam Tunnel Blowout Panel," was approved by PORC on January 2, 1997. The inspector reviewed BMO 96-18 and concluded that it adequately demonstrated that there was reasonable assurance that a HPCI HELB would not cause the EQ temperature limit to be exceeded, given the restriction on reactor building average temperature.

c. Conclusions

Prompt action was taken to verify the safety of continued operations when a HELB calculation, using conservative assumptions, determined that the EQ temperature limit for electrical equipment in the reactor building might be exceeded. The inspector will review the licensee's final corrective action for this condition in a future inspection. **(IFI 50-271/96-11-02)**

E2.3 Review of Temporary Modification No. 96-047, MS-V77 Leak Repair

a. Inspection Scope (37550)

The inspector reviewed the body-to-bonnet seal leak repair of MS-V77, main steam line drain outside containment isolation valve, per temporary modification (TM) No. 96-047, which involved the use of a Furmanite sealant injection process.

b. Observations and Findings

MS-V77 is a safety class 1, 3-inch pressure sealed, Walworth, gate valve. The initial attempt to repair the body-to-bonnet seal steam leak, via a sealant injection through two adapters drilled into the valve body, failed to completely stop the leak. The second attempt to close the leak utilized a different type sealant with greater ability to permeate the tight tolerance areas. Two additional holes were drilled and the injection amount was closely monitored using infra-red thermography to show the location and depth of the sealant injected. This helped to ensure that excessive amounts of sealant would not be injected into the fluid stream.

The inspector reviewed the injection procedure and found it consistent with EPRI guidelines for leak sealing. The inspector reviewed the 10 CFR 50.59 evaluation of the repair and found it consistent with regulatory requirements. (Also reference Section M1.3 of this report.)

c. Conclusions

The licensee implemented an acceptable procedure for temporary repair of valve MS-V77 until a permanent repair or replacement of the valve can be effected. The temporary modification controlling this repair effort was consistent with regulatory requirements for leak sealing using non-metallic compounds for temporary cessation of leakage.

E5 Engineering Staff Training and Qualification

E5.1 Curriculum Training

a. Inspection Scope (37550)

The inspectors observed a session of scheduled quarterly curriculum training provided to VY engineering personnel responsible for evaluating the operability of safety significant plant structures, systems, or components. The purpose of this observation was to evaluate the effectiveness of the topical training for providing management expectations and requirements regarding operability evaluations.

b. Observations and Findings

The inspectors found that engineering department managers were responsible for selecting and presenting the topical training at curriculum training sessions. During this training session the previous performance engineering manager (prior to the September 1996 engineering reorganization) conducted the session. The inspectors determined that the guidance provided to the engineers was very good. Specifically, the expectations for timely and qualitative technical assessments were clearly articulated and appeared to be well understood by the class following open discussions. Recent BMO process changes were presented including a revised engineering guideline document.

c. Conclusions

The BMO curriculum training was effectively conducted. Valuable information was presented to enable engineers to appropriately assess plant deficiencies and perform operability evaluations per established procedures.

E6 Engineering Organization and Administration

E6.1 Major Projects List

a. Inspection Scope (37550)

The inspectors reviewed the prioritization process established by VY for the resolution of technical issues requiring plant modifications and subsequently engineering support. This review assessed the established schedule for completion of such issues.

b. Observations and Findings

The inspectors found that the major projects work list was maintained by the plant manager and captured the most significant technical plant issues identified requiring resolution. This list included departmental issues including operator workarounds and engineering work including modifications and extensive assessments. The inspectors found that no formal acceptance criteria existed for adding, deleting, or limiting issues to the work list. Instead, a general consensus between department managers and the plant manager determined the prioritization of plant issues. Discussions with the plant manager and engineering personnel including managers revealed that recent actions had been initiated by VY to enhance the

administrative process and coordination of issues for inclusion and/or removal from the work list. These changes included the development of a plant procedure establishing weighing factors for issues and inclusion of such issues in the budget process.

The inspectors found that the work list established had been organized and planned through 1999 with responsibilities assigned to various departments for completion. Based on a review of the project topics and estimated completion dates, the inspectors did not identify any concerns with the established schedule.

c. Conclusions

The engineering work needed to support identified plant issues was appropriately included on the major project work list and engineering personnel knew the established schedule. The inspectors noted that VY had initiated actions to enhance the major project work list acceptance criteria for inclusion and prioritization of work.

E6.2 Performance Engineer Program Status

a. Inspection Scope (37550)

The inspectors performed a review of the status of VY's efforts to develop dedicated engineers for performance monitoring of safety-related or risk significant plant systems.

b. Observations and Findings

The inspectors found that the performance engineering department had recently redefined engineer responsibilities within the department to be dedicated as either plant support or system performance monitoring. At the time of this review, 11 performance engineers had been assigned to monitor 31 risk significant systems. Three vacancies remained to be filled: one position for a performance monitoring engineer and two positions for dedicated plant support engineers (for a total of four plant support engineers). The duties for plant support personnel included resolution of emergent issues, disposition of minor modifications, drawing updates, and setpoint changes.

The inspectors found that most performance monitoring engineers were currently pursuing training for qualification while supporting the design basis documentation reconstitution effort discussed in report section E8.1. VY's goal for developing a formal function-based performance monitoring plan for engineers was to be complete by mid-January 1997. VY's goal for maintaining qualification of all performance engineers was to be complete by August 1997.

c. Conclusions

VY continued to develop performance engineers for monitoring system performance.

E7 Quality Assurance in Engineering Activities

E7.1 Audit Report No. NSD-96-02

a. Inspection Scope (37550)

The inspectors reviewed quality assurance (QA) audit report no. NSD-96-02 to assess VY's QA involvement in engineering activities.

b. Observations and Findings

The inspectors found that this audit reviewed the effectiveness of engineering activities performed by both Design Engineering and the onsite engineering departments. The scope of this audit was thorough and supported by an in-depth review of documents and processes used by engineering personnel. Audit findings were well-supported and appropriately identified issues requiring engineering management attention.

The inspectors discussed the audit, audit process, and planned changes within the process with the QA Manager. The inspectors were informed that increased QA attention has been dedicated to address engineering programs, including the environmental qualification and IST (in-service testing) programs, based on previously identified engineering program issues at VY. Specifically, the number of QA audits have increased over the past year from 18 audits assessing engineering attributes to 23. Efforts have included expanded audit scopes and increased numbers of QA follow-up evaluations of departmental functional area self-assessments. In addition, VY has formed a Joint Quality Audit Group to better assess issues identified at other YNSD associated plants, including Maine Yankee. Another goal of this Joint Quality Group was to develop a standard audit type activity plan to be used when conducting QA reviews at various organizations.

c. Conclusions

The QA reviews were very valuable for identifying important engineering issues requiring management review. The inspectors determined that the audit scope was thorough and findings were well-supported. The inspectors noted VY's efforts to assess other YNSD associated plant issues via the establishment of the Joint Quality Audit Group.

E8 Miscellaneous Engineering Issues

E8.1 Design Basis Documentation Program Status

a. Inspection Scope (37550)

The inspectors reviewed the progress of the design basis document (DBD) program for validation of the plant design bases to ensure compliance of the as-built plant with regulatory requirements.

b. Observations and Findings

The inspectors discussed the status of the licensee's efforts with the Project Manager for the design basis effort. The inspectors found that the licensee had completed the "Design Basis Document Format and Writers Guide, " Revision 0, and developed technical guidance to ensure that consistent methodologies and levels of review were performed and corrective actions taken to ensure regulatory compliance.

The inspectors found that the validation process guidance established by VY presented the vertical slice assessment approach to independently evaluate system design and operability. This vertical slice assessment approach has been found appropriate by the NRC and utilized by the NRC during several nationwide inspections as described in NRC inspection procedures 93801, "Safety System Functional Inspection (SSFI)," 40500, "Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems," and 93808, "Integrated Performance Assessment Process." In addition, the inspectors found that guidance had been established to ensure acceptance criteria are appropriate for test procedures and also included a validation of design parameters.

At the time of this inspection, three of four initial DBDs prepared by one of the two different contract firms were in draft form. Based on VY's review of these four documents and the completion of four other documents being completed by the other contractor, the licensee planned to initiate a bid process for the remaining 15 DBDs outstanding. The 23 planned DBDs included those systems categorized as either high or medium risk significance in VY's Individual Plant Evaluation (IPE) report.

c. Conclusion

VY's commitment to develop and document safety-significant design bases for critical plant systems continued to progress. The methodologies established for documenting and validating design bases for plant systems were found to be consistent with NRC guidance.

E8.2 Licensee Event Report Reviews

a. Inspection Scope (92700)

Using the guidance of Inspection Procedure 92700, the inspectors reviewed the Licensee Event Reports (LERs) discussed below to verify the VY staff had implemented the corrective actions, as stated in the LERs, and to determine whether their response was appropriate and met regulatory requirements.

b. Observations, Findings, and Conclusions

- LER 96-014, Failure to provide tornado protection for diesel generator rooms as specified in the Final Safety Analysis Report due to unknown cause, dated June 21, 1996.

  This event and the current state of the licensee's corrective actions are discussed in section E2.2 of this report. The inspector reviewed LER 96-014 and assessed that the quality and timeliness of this report was good. The inspector noted that the

report stated that a supplemental LER, addressing results of the root cause investigation and all corrective actions taken, was to be submitted. Review of the supplement to this LER will be performed as part of the inspector follow-up of **IFI 96-11-01**. Accordingly, LER 96-014 is closed.

- LER 96-001, Supplement 1, Technical specification 4.6.E not met due to components not included in the inservice test program scope, dated July 22, 1996.

  LER 96-001 was reviewed in inspection report 50-271/96-03. In that review, the inspector also reviewed the licensee's Basis for Maintaining Operation (BMO 95-07, revision 1), and noted that the LER did not discuss valves that, although included in the IST program, were identified as not having been adequately tested. Supplement 1 to LER 96-001 documented these additional valves, including the specific testing deficiencies, operability considerations, and corrective actions. During the original LER review, the inspector had concluded that the licensee's operability determinations were adequately founded and that the interim and long-term corrective actions to resolve the testing inadequacies were appropriate. The licensee has committed to provide another LER supplement listing all IST program discrepancies identified during their comprehensive program review and corrective actions taken. LER 96-001 Supplement 1 is closed.

- LER 96-004, Supplement 1, Discrepancies identified in the Appendix J leak rate testing program, dated December 10, 1996.

  The original LER was reviewed in inspection report 50-271/96-03. Supplement 1 to the LER was submitted to document the root causes and corrective actions. The root causes were determined to be: (1) in the case of the reactor building closed cooling water (RBCCW) penetrations, a design analysis problem, in that a design change that downgraded the seismic classification of the RBCCW system did not identify the Appendix J program as a design reference or an attribute for the RBCCW system; (2) in the case of the core spray injection line penetrations, an administrative control problem, in that a commitment to submit an exemption was not tracked and, as a result, was not accomplished; and (3) in the case of the inboard main steam isolation valve (MSIV) test method, an administrative control problem, in that a commitment was not generated to review an NRC safety evaluation report (SER) to assure that the SER requirements were satisfied. The inspector considered that the case-specific and general corrective actions presented in this supplement were appropriate. LER 96-004, Supplement 1 is closed.

IV. Plant Support

R1 Radiological Controls

The licensee's program for occupational radiation exposure and radiation safety was reviewed. Specific areas reviewed included: audits and assessments; changes in personnel, procedures, and equipment; external exposure controls; internal exposure controls; dose to the embryo/fetus of declared pregnant women; control of radioactive materials and contamination, surveys and monitoring; and the program for maintaining personnel radiation exposures ALARA. The inspection also included a review of implementation of the UFSAR.

R3 Radiation Protection Procedures and Documentation

R3.1 External Exposure Controls

a. Inspection Scope (83750)

The inspectors reviewed the implementation of external exposure controls through tours of the facility, reviews of documentation, and interviews with various licensee individuals. The controls included posting and barriers for high radiation and very high radiation areas, posting of current plant status and dose information, ALARA caution signs, and use of the personnel dosimetry program. The inspectors made observations during tours of the facility and discussed radiological posting practices with technicians and supervisors.

b. Observations and Findings

Postings and barriers were appropriately used throughout the facility to warn workers of high radiation areas and prevent inadvertent entries with the exception of one area in the turbine building. The inspectors found a rope barrier on the floor that was to be used to delineate a high radiation area around the southeast section of the turbine (272-foot elevation). The rope barrier was supported through stanchions at various distances (approximately 10 - 20 feet) around the high radiation area. When observed by the inspectors on December 10, 1996, the rope was at waist-height in some areas, but on the floor between the stanchions in at least two other areas. At least one "High Radiation Area" sign was on the floor and was not legible. According to a radiological survey performed by the licensee's staff on November 15, 1996, dose rates were up to 150 millirem per hour in areas accessible from the rope boundary. The inspectors did not observe anyone in the area, and reported the problem to the Radiation Protection Supervisor.

The licensee's initial corrective actions included replacing the rope boundary, checking personnel entries into the turbine building, and touring the plant for other problems with high radiation area boundaries. The licensee's investigation revealed that all other areas were appropriately barricaded, no workers received an unexpected exposure during the previous 24 hours, but one worker was moving equipment on the turbine building floor (272-foot elevation) about 2 hours prior to the inspector's tour of the area. The worker had inadvertently knocked down the rope and stanchions with the equipment. The worker had made an attempt to replace the boundary, but did not do a sufficient job to maintain the integrity of the barrier, and did not report the problem to radiation protection personnel. The worker was subsequently reprimanded on his failure to report the incident. In addition, the Radiation Protection Manager had planned to write a short description, to be distributed to all employees, describing the event and proper conduct expected by Vermont Yankee management. This failure to follow licensee procedures and Technical Specifications had low safety significance, the appropriate corrective actions were taken by the licensee's staff, and the event was an isolated occurrence in an overall effective program to control external radiation exposures. This failure constitutes a violation of minor safety significance and is being treated as a Non-Cited Violation, consistent with Section IV of the NRC Enforcement Policy.

All areas with dose rates above 1000 millirem per hour were locked and controlled by radiological controls personnel to prevent unauthorized entry. Very high radiation areas (i.e., the TIP room) were locked and required administrative approval prior to entry.

Vermont Yankee sent all thermoluminescent dosimeters (TLDs) to the Yankee Atomic Laboratory for processing. The licensee's personnel dosimetry system had passed the laboratory testing performed by the National Voluntary Laboratory Accreditation Program (NVLAP). The current NVLAP certificate stated the accreditation for all categories was effective until September 30, 1997. The laboratory testing and onsite assessment are required every two years for NVLAP accreditation. NVLAP accreditation is required by the NRC for personnel TLD systems.

Personnel were observed by the inspectors wearing the appropriate dosimetry. The licensee used radiation work permits (RWPs) to control exposure and other radiological conditions. An automated access control system that interfaced with the alarming digital dosimeters was also used to track and control exposures. In a previous NRC Inspection Report (50-0271/96-09), the inspectors observed that the dose recorded from the digital dosimeters was actually lower (20 - 30%) than the dose assigned from a TLD for the same monitoring period. The lower readings were identified by the licensee's staff after some enhancements were made to the system software. Previously, the TLD dose was typically lower than the digital dosimeters and the licensee's staff was trying to eliminate this bias. This was identified as a program weakness by the inspectors since the digital dosimeter dose was used to control personnel exposure to avoid exceeding a regulatory or administrative dose limit. Even though the potential for exceeding a dose limit was low due to other administrative controls, this was identified as a program weakness. The licensee's staff was still evaluating the causes and effects of recent changes, as well as potential solutions to improve the precision of the dose tracking system.

The licensee's staff had recently implemented corrective actions to improve the bias of the electronic dosimetry system. These actions included recalibration of all electronic dosimeters to make the response closer to the response from the TLD. The inspectors reviewed the licensee's actions and discussed the changes with the licensee's staff. As of December 1, 1996, the electronic dosimeter and TLD doses were much more similar for the same monitoring period. The licensee's staff reported that the changes appeared to be effective, but there were not a large number of TLDs processed since the corrective actions were implemented. The staff believed the results at the end of the calendar quarter, when all personnel TLDs were processed, would provide more insight on the success of the changes.

Current survey information was posted at the main entrance to the RCA. The posted surveys were current and had sufficient level of detail to properly inform workers regarding radiation dose rates and contamination levels in work areas.

The inspectors reviewed records of personnel exposures for individuals who were required to be monitored as per licensee procedures and NRC regulations, including minors and declared pregnant women. There were no minors or declared pregnant women currently working in radiologically controlled areas at the facility. The records were maintained in good order and no deficiencies were identified.

c. Conclusions

In summary, the licensee provided very good controls for external exposure. High radiation areas were appropriately barricaded and/or locked with the exception of one high radiation area in the turbine building that was identified by the inspector as a non-cited violation. External dosimetry was appropriately used by licensee personnel. Radiological work was well controlled through the use of RWPs and the automated access control system. Current radiological and plant information for work areas were available to workers. The licensee provided good radiological briefings for drywell workers. Minor program weakness was previously identified with the dose assignment process from the electronic dosimeter system, but recent changes have attempted to improve the bias between TLD and electronic dosimeter response. Records of external exposure and total occupational exposure were maintained as appropriate.

R3.2 Internal Exposure Controls

a. Inspection Scope (83750)

The inspectors reviewed the implementation of internal exposure controls through tours of the facility, reviews of documentation, and interviews with various licensee individuals. Controls included air handling systems or other engineering controls for potential airborne areas, air monitoring, posting of airborne areas, use of respiratory protection, and internal dose assessments.

b. Observations and Findings

The inspectors verified through tours that air monitoring and ventilation equipment was available for use in potential airborne areas. Although no areas were posted as an airborne area, HEPA filters were provided for decontamination areas. There was only one respirator issued for radiological control purposes in 1996 (during the refueling outage). This low use of respiratory protection was a result of successful controls to limit the concentrations of airborne radioactivity in the work areas.

The inspectors reviewed internal dose assessments and assignments during 1996. The assignments were derived through a calculation based on the workers' stay times in an area and measured concentrations of airborne radioactivity, as well as bioassay (whole body counting) measurements.

During 1996, the highest individual committed effective dose equivalent (CEDE) assignment was 26 millirem and was assigned during the refueling outage. All internal dose assignments since the last inspection of this area were less than 10 millirem. NRC regulations allow a total occupational dose assignment (internal plus external dose) up to 5000 millirem per year. The dose assessments were derived using the licensee's approved procedures and technically acceptable assumptions.

Also reviewed were the records and reports required after the workers had terminated employment at the site. The reports were in compliance with 10 CFR 20 and licensee procedures.

c. Conclusions

Overall, the license effectively provided internal exposure controls. Engineering controls were made available to help limit airborne radioactivity and internal intakes. Air monitoring was performed to assess the concentration of radionuclides in the work area. Internal dose assignments were very low as compared to regulatory limits and total assigned dose. Records and reports of internal dose assignments were thorough and well documented.

The inspectors determined that the licensee demonstrated very good contamination controls during the outage and provided very good air sample monitoring program with the result of very low internal exposures to personnel.

R3.3 Control of Radioactive Materials and Contamination, Surveys and Monitoring

a. Inspection Scope (83750)

The inspectors reviewed the implementation of the controls of radioactive materials and contamination, surveys and monitoring through tours of the facility, reviews of documentation, and interviews with various licensee individuals. These controls included posting and identification of contaminated areas, labelling of radioactive materials, monitoring and frisking equipment, use of protective clothing, and radiological surveys of work areas.

b. Observations and Findings

Control of radioactive materials was good throughout the radiologically controlled areas (RCAs) of the plant. Caution labels were attached to bags of trash, storage bins and cabinets, vacuums, and miscellaneous equipment. All items with caution labels were also labelled with other appropriate information such as dose rates, contents, and date of survey.

Contamination controls were generally good. Decontamination of areas after the refueling outage was generally very successful. Very few areas that required routine personnel access were controlled for contamination purposes. Contaminated areas were posted with warning signs and delineated with rope boundaries. Step-off pads were provided to mark entry and exit points from the contaminated areas. Containers were provided for potentially contaminated protective clothing at the exits from the contaminated areas. Workers in contaminated areas were observed wearing the appropriate protective clothing. Current contamination survey information was posted at the main entrance to the RCA.

The licensee was challenged by a steam leak in the torus room of the reactor building (MS-77 valve) in early December. The inspectors observed the area with steam leaking on various days, and noted that the amount of steam escaping from the valve was increasing. The immediate response by the licensee's radiation protection staff included placing a rope boundary and a "contaminated area" sign in a small area around the leaking valve (approximately 4 feet by 8 feet in area). A small catch container was used to collect some of the steam/water with a hose leading to a floor drain on the lower elevation. As the leak increased, the catch container was not large enough, the steam condensed and water dripped down around the torus outer surface to the floor of the lower elevations of the torus

room. The inspectors noted puddles of water on the torus room floor that were also surrounded by rope boundaries and posted with "contaminated area" signs. However, as the leak progressed, the water on the torus room floor started to move toward the contaminated area boundary into a walkway used by various personnel. The inspectors noted that the radiation protection staff had not taken any further action to isolate the water or direct it to a floor drain. The inspectors reported this condition to the radiation protection staff and the staff responded by placing absorbent materials at the edge of the contaminated area on the floor, and by enlarging the contaminated area on the top of the outside of the torus. The inspectors found that the licensee's staff was slow to respond to the leak and resulting contamination. The safety significance of the potential personnel contaminations or spread of the contaminated area was relatively low, but the slow response in identifying an adverse radiological condition was considered an unusual performance in an usually responsive organization.

Monitoring and frisking equipment were available throughout the plant at various locations. The inspectors verified through random reviews that the equipment in the field was within the current calibration period and had documented daily performance checks.

c. Conclusions

The licensee provided very good controls for radioactive materials. Contamination controls were generally good. The licensee's radiation protection staff was challenged by a steam leak in the torus room and the staff's response to isolating and containing the contamination was slower than usual. Caution labels and postings were used appropriately. Many areas were decontaminated after the refueling outage to allow easier access by plant employees. Equipment for monitoring and control of contamination was maintained and made available to workers throughout the facility.

R3.4 ALARA Program

a. Inspection Scope (83750)

The inspectors reviewed the implementation of the ALARA program through tours of the facility, reviews of documentation, and interviews with various licensee individuals.

b. Observations and Findings

The licensee had established a goal for less than 180 person-rem for work activities performed during the refueling outage. The actual total exposure to all workers was approximately 176 person-rem as of November 2, 1996. The actual exposure was less than the exposure accrued in previous outages. The licensee attributed this lower exposure total to program initiatives including various dose reduction ideas and lower dose rates in most areas of the plant. The licensee's total exposure goal for the year (1996) was 220 person-rem. As of December 8, 1996, the site personnel had accumulated approximately 225 person-rem. Although this total was higher than the original goal, the licensee's staff had achieved very good results for the amount and scope of radiological work that was accomplished during 1996. The highest individual total dose for the year was less than 2 rem. The annual occupational exposure regulatory limit is 5 rem.

Some of the exposure reduction initiatives used during the refueling outage included the use of cameras for firewatch in the drywell, trial use of a radio digital dosimeter system, increased shielding, system flushes, and the use of mock-ups or pictures of components for pre-job briefings.

The inspectors found that the ALARA reviews were well documented, including post-job reviews and active reviews of on-going work. The active reviews of on-going work were not always filed with the ALARA review documents as noted in a previous inspection, but the organization of the documentation had improved after the outage. The post-job reviews were performed by various technicians who had been assigned to the specific jobs during the outage. The resulting reviews were of mixed quality, but presented the direct experience from the technicians and generally contained good, detailed descriptions. The reviews included areas that could be improved to further lower workers' radiation exposures during the next performance of the various tasks/jobs.

c. Conclusions

The licensee's staff had established aggressive and realistic personnel radiation exposure goals for 1996. Dose reduction initiatives were implemented for the refueling outage activities. The inspectors determined that the licensee's staff was successfully using the information to continue to improve processes and procedures and lower workers' radiation exposures.

R4 Staff Knowledge and Performance in Radiation Protection and Chemistry

R4.1 Annual PASS Sample Exercise

a. Inspection Scope (83750)

The inspectors observed the performance of the radiation protection staff during the annual PASS sample exercise and the conduct of a licensee critique following the exercise.

b. Observations and Findings

The licensee's staff conducted an emergency exercise on December 10, 1996, with the objective of obtaining a post-accident sampling system (PASS) sample from the reactor coolant for laboratory analysis. Two technicians worked together to obtain the sample at the PASS sampling panel in the auxiliary building.

The inspectors observed the briefing for the radiation protection technicians prior to obtaining the sample. The briefing included the dose rates in the area, radiological concerns, and dosimetry and protective clothing requirements. The inspectors noted good use of the procedure during the sample collection. One technician read aloud the procedure, while the second technician repeated the instruction for verification, waited for verification from the first technician, and then performed the process. The procedure was very detailed and included requirements to note vacuum pressure readings and dose rates at various times in the process.

The inspectors observed very good radiological control practices by the technicians including continuous dose rate monitoring, communication with the main radiological controls checkpoint, proper use of protective clothing, and appropriate handling of radioactive materials (including contamination controls).

The sample was transferred to chemistry personnel for sample analysis and reporting of results. The inspectors did not observe the chemistry analysis portion of the exercise, but the analysis was actually performed within the 3 hour required time period.

The exercise participants held a critique meeting after the exercise to discuss the various aspects of the exercise. The participants feedback was very detailed and was documented for future actions, including improvements to the process, procedure, and conduct of the exercise. The inspectors found that the critique meeting was very useful and the suggestions for improving the timeliness of PASS samples was well founded.

c. <u>Conclusions</u>

The licensee performed very well during the annual emergency exercise for obtaining a reactor coolant PASS sample. The technicians followed the licensee's procedure and no major problems were noted. Actual PASS sample operation was very good experience and thoroughly tested the quality of the written procedure. The post-drill critique was useful and identified several areas for improvement.

R4.2 Radiological Controls During Calibration of Off-Gas Monitor at the Plant Stack

a. <u>Inspection Scope (83750)</u>

The inspectors observed the radiation protection staff coordinate and perform the calibration of the accident range on the off-gas radiation monitor at the plant stack.

b. <u>Observations and Findings</u>

The radiation protection staff performed a calibration on the accident range of the off-gas monitor using a radiography source from another company licensed by the NRC for possession and use of the source. The radiography source in the unshielded position yielded high radiation dose rates and required careful controls. The Vermont Yankee staff used an approved radiography procedure to perform the calibration. The procedure included appropriate radiological controls such as posting of high radiation areas, barricades to prevent unauthorized entry, dose rate surveys, and communications with the control room. The inspectors found that the licensee's staff appropriately implemented the procedure and provided very good radiological controls. Specifically, the controls observed included appropriate barriers and postings, proper radiation surveys around the perimeter of the plant stack during irradiations, monitoring of the dose rates at the property fence, and constant communications with an individual in the control room.

c. <u>Conclusions</u>

The licensee's staff provided very effective radiological controls during the use of a radiography source used for calibration of the off-gas radiation monitor at the plant stack.

R6 Radiation Protection Organization and Administration

R6.1 Changes in the Radiological Controls Program

a. Inspection Scope (83750)

Changes to the radiation protection program were reviewed by the inspectors through interviews with licensee personnel.

b. Observations and Findings

No major changes were made to the radiation protection program. The licensee had made some temporary changes in the radiation protection organization for the refueling and maintenance outage, but the organization reverted to the previous size and responsibilities at the end of the refueling outage.

c. Conclusions

The organization was effective and no negative consequences from the temporary changes were observed.

R7 Quality Assurance in Radiation Protection Activities

R7.1 Audits and Appraisals

a. Inspection Scope (83750)

Audits, surveillance reports, and internal assessments of the radiological controls program conducted since the last NRC inspection were reviewed by the inspectors.

b. Observations and Findings

There had been one formal quality assurance (QA) audit of the radiological controls program conducted since the last NRC inspection of program. The audit was conducted in May 1996. The auditors reported that the radiation protection program was effective. Several program strengths were identified including documented daily tours of radiological areas by the radiation protection staff, radiological housekeeping throughout the plant, and procedural control of portable air filtration equipment. No program weaknesses were identified, but three recommendations for improvement were listed. The inspectors reviewed the audit report and noted that the report had been reviewed by appropriate management personnel.

The licensee's QA staff had performed various surveillances of radiological control activities during the outage. The activities under surveillance included pre-job briefings for radiological activities, personnel exit from contaminated areas, posting and labelling of radioactive materials and contaminated items, observation of torus divers, high radiation area entries, ALARA package documentation review, radiological survey documentation review, radiological systems breach, general radiological control practices, and reactor

disassembly and inspection. The surveillance reports documented minor problems. In response, the licensee initiated timely and technically acceptable corrective actions to prevent recurrence in all responses reviewed by the inspectors.

c. Conclusions

The inspectors concluded that the licensee continues to improve the quality of the radiological controls program through the self-identification and correction of minor deficiencies and program areas for improvement. No violations of NRC regulations or major safety concerns were identified.

R8 Miscellaneous Radiation Protection Issues

R8.1 Review of UFSAR Commitments

a. Inspection Scope (83750)

The inspectors reviewed the implementation of the Updated Final Safety Analysis Report (UFSAR) in the area of radiological controls (Section 13.3). Specific items included use of personnel monitoring, personnel access control, and radiation instrumentation.

b. Observations and Findings

A recent discovery of a licensee operating its facility in a manner contrary to the UFSAR description highlighted the need for a special focused review that compares plant practices, procedures and/or parameters to the UFSAR description. While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the UFSAR that related to the areas inspected. No inconsistencies were noted between the wording of the UFSAR and the plant practices, procedures, and/or parameters observed by the inspectors. However, one statement was not very clear to the inspectors. Section 13.4.1.6, titled "Health Physics Instrumentation" stated that, "Portable radiation monitoring instruments are available to control room personnel to measure radiation dose rates and airborne concentrations within the control room." The inspectors asked the licensee's representatives what the intent of this statement was relative to actual practice. The representative stated that the intent of the statement was to describe that portable radiological instrumentation was available to monitor the control room conditions in the control room as opposed to meaning that the instrumentation was available for use by the control room personnel. The representative further stated that control room personnel were not qualified to operate portable radiation monitoring instruments. The inspectors verified that portable radiation monitoring instruments were not located in the control room; however, they were available to qualified individuals in the radiation protection department. Subsequently, the inspectors verified that the licensee submitted a written request to change the wording in the UFSAR to read as follows, "Portable radiation monitoring instruments are available onsite to measure control room radiation dose rates and airborne concentrations." The inspectors agreed that this change better described the actual practice; however, the inspectors must further review licensee documentation to ensure that the intent of this change does not conflict with other licensee commitments **(URI 96-11-03)**.

c. Conclusions

In summary, the licensee is effectively implementing the UFSAR commitments. No inconsistencies were identified in the radiation protection program activities as described in the UFSAR. An unresolved item was identified regarding the licensee's initiation of a change to one section of the UFSAR.

R8.2 LER No. 96-29 Review

a. Inspection Scope (71750)

Licensee Event Report No. 96-29, Process and communications inadequacies result in the failure to analyze emergency diesel generator (EDG) fuel oil within time allotted by TS surveillance requirement, dated January 3, 1996, as the title states, identifies a problem with the analysis of an EDG fuel oil sample. The inspector reviewed the quality of the LER, the facts involving this problem, and the licensee's root cause and corrective actions to assess the appropriateness of the VY staff's actions.

b. Observations and Findings

Based upon a detailed followup of the event, discussion of the problem with the responsible plant staff, and observation of the Plant Operations Review Committee (PORC) deliberations of this issue, the inspector identified that the VY staff had appropriately taken EDG fuel oil samples at the TS prescribed interval of 30 days. As stated in the LER, the November 22, 1996 sample was not analyzed until December 9, 1996. The VY staff viewed this as a TS non-compliance because the previous EDG sample (taken on October 18) analysis results were obtained on October 30. Therefore, the time between VY staff receipt of the EDG fuel oil sample analyses exceeded 30 days (October 30 to December 9 was 41 days).

The inspector re-examined the TS requirement (TS 4.10.C.2) which states that "once a month a sample of diesel fuel shall be taken and checked for quality," and discussed the intent of this TS with the NRR TS Branch staff. Based upon this review, the inspector determined that this TS requires the fuel oil to be sampled every 30 days and implies that the sample should be analyzed prior to the next 30-day sample being taken. There is no imposition of the same 30-day time interval between receipt of sample analyses.

Consequently, the inspector concluded that the VY staff did not violate TS 4.10.C.2 for EDG fuel oil sampling. The inspector did verify that satisfactory EDG fuel oil sample results were obtained for the December 20, 1996 sample and noted that the analysis results were received by the VY staff from their vendor on January 6, 1997. The inspector also observed that the corrective actions for the peripheral issues (poor VY staff communications and mishandling of the sample shipment) appeared to be appropriate and were effective in ensuring the timely analysis of the December 20 sample.

c. Conclusion

The VY staff did not violate the Technical Specifications for periodic sampling and analysis of EDG fuel oil as stated in LER No. 96-29, dated January 3, 1997. Notwithstanding, their corrective actions to address the peripheral VY staff performance concerns were observed to be appropriate and effective.

F8 Miscellaneous Fire Protection Issues (71750, 92904)

F8.1 Licensee Event Report Reviews

a. Inspection Scope (92700)

Using the guidance of Inspection Procedure 92700, the inspectors reviewed the Licensee Event Reports (LERs) discussed below to verify the VY staff had implemented the corrective actions, as stated in the LERs, and to determine whether their response was appropriate and met regulatory requirements.

b. Observations, Findings, and Conclusions

- LER 96-020, Inadequate vendor design activity and licensee design verification result in inability to demonstrate fire suppression system operability, dated October 7, 1996.

  The issue of switchgear rooms' fire suppression system operability was first reviewed in inspection report 50-271/96-03, and was the subject of extensive review in inspection report 50-271/96-08. Design changes to address testing deficiencies are being tracked as unresolved item 96-08-01. Based on previous reviews, this LER is closed.

- LER 96-007, Supplement 1, Vital fire dampers not installed in accordance with manufacturers instructions, dated December 18, 1996.

  The original LER was reviewed in inspection report 50-271/96-06. Supplement 1 to the LER was submitted to document the root cause of the event. The cause was determined to be human error, in that the manufacturers instructions were not followed during installation. Long term corrective action indicated that, of the dampers at issue, seven were repaired or replaced, and four were determined to be acceptable as-is. LER 96-007, Supplement 1 is closed.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors met with licensee representatives periodically throughout the inspection and following the conclusion of the inspection. At that time, the purpose and scope of the inspection were reviewed, and the preliminary findings were presented. The licensee acknowledged the preliminary inspection findings.

X3 Management Meeting Summary

X3.1 Commissioner Rogers Visit to Vermont Yankee

On December 19, Commissioner Kenneth C. Rogers toured the facility and met with VY management personnel. Commissioner Rogers also met with the Vermont State Nuclear Engineer, Mr. William Sherman.

X4 Licensee Management Changes

On January 10, Vermont Yankee announced the following management changes:

- The current Plant Manager, Bob Wanczyk, was appointed to a newly created position, Director of Safety and Regulatory Affairs. In this capacity, he will oversee all quality assurance and licensing activities, and will report to the Vice President - Operations. Mr. Wanczyk will also assume the role of Nuclear Safety and Review Committee Chairman.

- Greg Maret, currently the Operations Superintendent, would assume the position of Plant Manager. This reassignment was effective on January 13.

- Effective January 29, Kevin Bronson will assume the position of Operations Manager from Larry Doane, who will be reassigned within the Operations department. Mr. Bronson has been with Vermont Yankee since 1981 and is currently an Operations department shift supervisor.

X5 Review of Updated Final Safety Analysis Report (UFSAR)

A recent discovery of a licensee operating its facility in a manner contrary to the UFSAR description highlighted the need for a special focused review that compares plant practices, procedures, and parameters to the UFSAR description. While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the UFSAR that related to the areas inspected. Discrepancies that were noted were documented in the applicable section of the above report.

INSPECTION PROCEDURES USED

71701	Plant Operations
37550	Engineering
37551	Onsite Engineering
61726	Surveillance Observations
62707	Maintenance Observations
71714	Cold Weather Preparations
71750	Plant Support Activities
92700	Onsite Followup of Written Reports of Non-routine Events
93702	Onsite Response to Events
83750	Occupational Radiation Exposure

ITEMS OPENED, CLOSED, AND DISCUSSED

Closed

IFI 96-09-05	Primary Containment Nitrogen Purge System Isolation Valve Leakage

Opened

IFI 96-11-01	Emergency Diesel Generator Tornado Protection
IFI 96-11-02	Follow-up of issue involving steam tunnel blowout panel actuation setpoint variance potentially impacting EQ assumptions.
URI 96-11-03	Follow-up on change to UFSAR to ensure it does not conflict with other licensee commitments.

PARTIAL LIST OF PERSONS CONTACTED

<u>Licensee</u>

*R. Wanczyk, Plant Manager
*G. Maret, Operations Superintendent
E. Lindamood, Director of Engineering
*L. Doane, Operations Manager
M. Watson, I&C Manager
*F. Helin, Reactor Engineering Manager
M. Desilets, Radiation Protection Manager
S. Skibniowsky, Chemistry Manager
G. Morgan, Security Manager
S. Jefferson, Asst. to Plant Manger
J. DeVincentis, Asst. to Director of Engineering
D. Calsyn, QA Supervisor, YAEC
J. Cox, Radiation Protection Supervisor
J. Geyster, Plant Health Physicist
P. Guido, Radiation Protection Supervisor
R. Morrissett, ALARA Engineer
M. Thornhill, Radition Protection Supervisor

* Effective January 13, 1997 these names and positions have changed

LIST OF ACRONYMS USED

VY	Vermont Yankee
TS	Technical Specifications
LER	Licensee Event Report
FSAR	Final Safety Analysis Report
HHB	house heating boiler
ER	Event Report
CW	circulating water
FPP	freeze protection panel
CS	core spray
LCO	limiting condition for operation
WO	work order
TM	temporary modification
BMO	basis for maintaining operability
MSIV	main steamline isolation valve
EDCR	engineering design change request
SSPV	scram solenoid pilot valves
SDV	Scram discharge valve
ASD	alternate shutdown
QA	Quality Assurance
IST	Inservice testing
DBD	design basis document
EDG	emergency diesel generator