ENCLOSURE 1

EXECUTIVE SUMMARY

Vermont Yankee Nuclear Power Station
NRC Inspection Report 50-271/96-03

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 6-week period of resident inspection.

Operations

Overall, the conduct of plant operations was professional and safety conscious this inspection period. The Plant Operations Review committee (PORC) exhibited a clear and well defined safety focus during their examination of recent Inservice Testing and Appendix J Program discrepancies and during their review of a proposal for alternate Appendix R compensatory measures. Prompt and effective actions were taken by the VY staff to resolve the operability concern involving the loose valve operators on the manual isolation valves to both residual heat removal heat exchangers.

Maintenance

A number of maintenance and testing activities were observed and found to be well coordinated, with good pre-evolutionary briefings and good communications. Plant staff response to the March 26 recirculation pump trip was good, however, the apparent cause of the trip was identified to have been personnel error. An inspection follow-up item (IFI 96-03-01) was assigned to review VY's root cause evaluation and corrective actions.

The VY staff's approach to monitoring and understanding the scram solenoid pilot valve VITON diaphragm degradation issue has been and continues to be aggressive. However, VY's increased frequency of individual rod scram time testing may potentially conflict with Technical Specifications 4.3.C.2 if appropriate administrative controls are not instituted. Pending further VY staff and inspector review, this issue is unresolved (URI 96-03-02).

The VY staff's decision to postpone the reactor core isolation cooling system and the "B" emergency diesel generator (EDG) limiting condition for operation (LCO) maintenance outages, during this inspection period, demonstrated prudent decision making with safety benefits.

Engineering

Identification of the battery room masonry wall seismic qualification calculation errors demonstrated an excellent questioning attitude on the part of the individual engineer. The engineering and plant staff handling of this design non-conformance, with respect to promptly dispositioning the station batteries operability impact, was not timely. PORC's review of the station batteries operability determination was completed and, as referenced above, the PORC's decision to postpone the "B" EDG LCO maintenance outage was

prudent. The NRC staff review of this potentially degraded condition using the guidance of Generic Letter 91-18 was ongoing at the conclusion of the inspection period and was unresolved (URI 96-03-03).

VY engineering and operating staffs' have appropriately dealt with the torus water temperature limit concern, to date, by pursuing further design basis analyses and, in the interim, administratively restricting torus water temperature to 90 degrees F. Pending completion of formal analysis of this potential design basis conflict and NRC staff review, this issue is unresolved (URI 96-03-04).

Licensee identified and corrected discrepancies in the Inservice Testing and Appendix J Programs (reference LERs 96-001 and 90-004, respectively) were dispositioned as non-cited violations. These discrepancies were identified by the VY staff as a result of thorough corrective action for organizational problems identified via the Fire Protection and Appendix R Programs.

The inspector reviewed the current technical status of the advanced off-gas (AOG) system as part of the verification process for the licensee's letter (No. BVY 96-17), dated February 26, 1996. In particular, the inspector reviewed issues dealing with AOG system performance and with a system modification cancellation. The engineering staff's coordination with the plant staff, the quality of the consolidated as-built panel 9-50 electrical drawings, and the delineation and resolution of design issues for the planned AOG modification were generally very good. The inspector found no indication that the cancellation of the modification was driven by cost considerations other than the inherent cost risk associated with implementing a modification with possible incomplete documentation, such as installation and test instructions. The inspector found no engineering or maintenance indications in the last 5 years that AOG system functionality was impaired in such a manner that led to degraded conditions that exceeded the Technical Specification requirements. Recent initiatives including system-analyzed maintenance developed by the I&C engineering staff and reliability-based maintenance developed by the maintenance engineering staff were considered good.

Plant Support

VY's ongoing systematic re-examination of the entire Fire Protection and Appendix R Programs identified a number of improperly installed fire dampers and incomplete test data for the switchgear rooms carbon dioxide suppression systems. The compensatory measures for these discrepancies were promptly implemented and the proposed corrective actions deemed appropriate. Conclusive system test results to support a system operability determination are still pending and this issue remains unresolved **(URI 96-03-06)**.

VY staff review of plant refueling practices identified that preceding the 1990 and 1992 refuel outages all three layers of reactor vessel shield blocks were removed while at power. This condition was determined to have been in conflict with the plant design basis. The apparent root cause of this problem was inadequate procedural guidance, but further evaluation was ongoing. Pending VY completion and inspector review of the final root cause evaluation, this issue is unresolved (URI 96-03-05).

ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.	50-271
Licensee No.	DPR-28
Report No.	96-03
Licensee:	Vermont Yankee Nuclear Power Corporation
Facility:	Vermont Yankee Nuclear Power Station
Location:	Vernon, Vermont
Dates:	February 2 - March 30, 1996
Inspectors:	William A. Cook, Senior Resident Inspector
Approved by:	Richard J. Conte, Chief, Projects Branch 5 Division of Reactor Projects

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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.

On March 5, VY representatives met with the NRC staff in Headquarters at Rockville, Maryland to discuss issues related to the Fire Protection and 10 CFR 50 Appendix R Programs at the station. The VY staff provided status of their progress and plans for establishing compliance with Appendix R prior to the completion of the 1996 refueling outage.

On March 13, VY representatives met with the NRC staff in Headquarters to discuss the reactor vessel shroud repair and reactor vessel internals inspection plans. Details of the 1995 outage shroud visual inspection results and planned 1996 outage shroud repair plans were shared with the NRC reviewers and inspection staff. The formal repair plan and plant modification approval request has not been submitted for NRC staff review, to date.

On March 28, VY representatives met with the NRC staff in Headquarters to discuss the seismic qualification concerns involving the station 125 VDC battery room masonry block walls. The VY representatives provided additional information pertaining to the specific seismic qualification methodology being used to support their masonry block wall operability determination (reference Section 3.1.1 of this report).

1.0 OPERATIONS

1.1 Conduct of Operations

Using Inspection Procedure 71707, the inspector conducted frequent reviews of on-going plant operations. The conduct of operations, as observed in the control room and in the plant, was professional and safety conscious. Specific operating events and noteworthy inspector observations are detailed in the sections below.

1.2 Operational Status of Facilities and Equipment

1.2.1 Safety Parameter Display System (SPDS) Out of Service (71707, 93702)

On March 1 at 3:40 p.m., VY notified the headquarters duty officer in accordance with 10 CFR 50.72 (Event No. 30053) that the SPDS was out of service for greater than eight hours. The inspector determined that the Emergency Response Facility Information System (EFIS), which includes the SPDS, had been removed from service at 7:40 a.m. for preventive maintenance. An unanticipated problem with the data acquisition system delayed restoration of the system within the planned eight-hour work window. The ERFIS/SPDS was returned to service at 8:35 p.m. on March 1.

The inspector verified that compensatory measures to manually conduct thermal heat balance calculations were in-place, as well as, an alternate means to verify reactor core thermal limits, since the ERFIS provides these functions.

Neither back-up method was warranted because the ERFIS was restored in less than 24 hours. The inspector concluded that VY made the appropriate 10 CFR 50.72 notification for this temporary loss of emergency assessment capability and that appropriate compensatory measures were available.

Inspector follow-up of a related 10 CFR 50.72 notification (Event No. 29846) on January 17, identified that similarly appropriate compensatory measures were available, but not needed. The January 17 SPDS outage lasted eight hours and 12 minutes and was the result of a planned system outage to replace the plant process computer system. The inspector determined that unforeseeable minor software changes delayed restoration of the SPDS within eight hours on January 17.

1.2.2 Battery Room Block Wall 10 CFR 50.72 Notification (71707)

On March 18, VY made a 10 CFR 50.72 notification identifying that the battery room masonry block wall did not meet current design basis criteria for seismic qualification. Additional details and inspector observations are in Section 3.1.1.

1.2.3 Torus Water Temperature Design Limit Concern (71707)

On March 26, VY made a 10 CFR 50.72 notification identifying that the current Technical Specification (TS) torus water maximum temperature limit may be non-conservative. This determination was based upon a new primary containment loss of coolant accident (LOCA) response analysis. Additional details and inspector observations are in Section 3.2.1.

1.3 Operations Procedures and Documentation (71707)

1.3.1 Monthly Statistical Reports

The inspector reviewed the Monthly Statistical Reports for January and February 1996, dated February 10 and March 10, 1996, respectively. The inspector verified that these reports were submitted in accordance with TS 6.7.A.3 and properly reflected the operating status of the facility during the months of January and February 1996.

1.3.2 Fuel Failure Status and Parameter Trends Report

The inspector reviewed the internal monthly fuel failure status and trend reports for the months of December 1995, January 1996, and February 1996. These reports are generated by the site reactor engineering group to monitor and trend fuel performance by examining offgas radiation levels and reactor coolant activity analyses. Between December 1995 and February 1996 no fuel failures occurred and no abnormal trends were observed. The inspector found this plant performance monitoring appropriate and the summary reports well written and concise.

1.4 Quality Assurance in Operations

On Site Review Committee Activities

a. Inspection Scope

On February 28, the inspector observed routine Plant Operations Review Committee (PORC) meeting No. 96-019. The inspector observed the PORC review of two Licensee Event Reports (LERs 96-01 and 96-04) and Basis for Maintaining Operability (BMO) No. 95-07, Revision 1. These types of items obtain a PORC review prior to approval and issuance by the Plant Manager. At PORC meeting 96-019, the inspector also observed a discussion pertaining to proposed alternate compensatory measures for Appendix R issues.

b. Observations and Findings

The inspector observed the PORC members' review and discussions of LERs 96-01 and 96-04 and BMO 95-07, Rev. 1, to have been thorough and insightful. The committee ensured that questions raised, pertaining to BMO 95-07 by the members from the previous PORC meeting quorum, were appropriately addressed and resolved in Revision 1 to BMO 95-07. The inspector also noted that the PORC members asked some broader safety questions during their discussion of LER 96-04 which resulted in appropriate PORC followup items. Additional NRC observations of PORC are discussed in Sections 2.3.1 and 3.1.

The observed discussion of the proposed alternative Appendix R compensatory measures, presented by the Fire Protection Improvement Plan (FPIP) task force, demonstrated a clear understanding by the PORC members of their chartered responsibilities. Briefly, the PORC rejected the FPIP task force's request to review and comment on their alternative compensatory measures proposal. The PORC rejected the review of this proposal because it had not already been scrutinized through appropriate administrative procedural review and approval (including a 50.59 safety evaluation, if needed) processes.

The inspector considered PORC's refusal to engage in a discussion of the merits of the alternative Appendix R compensatory measures proposal without the conduct of prerequisite reviews as entirely appropriate.

c. Conclusion

The inspector concluded that the PORC appropriately implemented its charter and exhibited a clear and well defined safety focus during their review of the above stated agenda topics. The PORC's rejection of the FPIP task force proposal exemplified their high standards of procedural review and defense-in-depth through required administrative processes.

1.5 Miscellaneous Operations Issues (92700)

(Closed) LER 50-271/96-06: Potentially Inoperable Residual Heat Removal (RHR) Service Water Valves Due to the Bolts Holding the Valve Operators Being Insufficiently Tight, dated March 14, 1996

On February 14, VY made a 10 CFR 50.72 notification (Event No. 29975 identifying the entry into a TS required shutdown due to both RHR loops' heat exchanger manual outlet isolation valves (RHR-192A & B) being declared inoperable. Both RHR-192A & B being inoperable resulted in the containment cooling, RHR service water, and alternate cooling systems being declared inoperable at 10:15 a.m. VY declared the valves inoperable because the four blind bolts affixing the manual valve operators to the valve bonnets had become loose and, if sufficiently un-threaded, could have potentially prevented valve operation. The valve design is such that, if the operator became detached, the direction of flow through the valve could cause it to shut and stop cooling water flow to the heat exchanger.

The inspector observed that corrective maintenance was promptly initiated and each valve was removed from service, repaired, tested, and returned to service, one after the other. The sequencing of valve repairs ensured one train was available while repairs were effected on the other. Both trains were restored by 3:00 p.m. on February 14 and the TS shutdown terminated.

During valve repair, the blind bolts were found to be slightly loosened, but not sufficiently to prevent operation of the manual valve operator. The inspector concluded that VY took prompt and effective action to resolve this system operability issue upon identification of its potential safety impact. The corrective maintenance was well planned, executed, and controlled. The LER was well written and concise. VY committed to submit a supplement to LER 96-06 upon completion of the root cause evaluation.

2.0 MAINTENANCE

2.1 Conduct of Maintenance

2.1.1 Review of Maintenance and Surveillance Testing

a. Inspection Scope

Using Inspection Procedures 62703 and 61726, the inspector reviewed all or portions of the following maintenance and surveillance testing activities.

- Monthly surveillance testing of the A and B emergency diesel generators conducted on February 20.
- Post-maintenance testing of the B control rod drive pump conducted on February 23 and March 1.
- Control rod pattern exchange and single rod scram time testing conducted on February 27.
- High pressure coolant injection systems full flow testing conducted in accordance with procedure OP-4120 on March 7.
- Post-modification testing of recirculation loop sample valves V2-39 and V2-40 conducted on March 8.

b. Observations Findings and Conclusions

The inspector monitored the pre-evolutionary briefings conducted in the control room and found the briefings to be well structured and comprehensive.

Testing personnel clearly understood the test acceptance criteria and the step-by-step testing sequence. The inspector observed good communications and coordination of testing personnel activities with routine plant evolutions.

2.1.2 Recirculation Pump Trip Due to Maintenance Personnel Error (93702)

a. Background

On March 26 at 11:41 a.m., while operating at 100 percent reactor power, the "B" recirculation pump tripped. Operators appropriately responded to the recirculation pump trip and stabilized the unit at approximately 50 percent power. The "A" recirculation pump was manually ramped back and control rods were manually inserted to exit the Buffer Region of the TS Section 6.7.A.4 established Core Operating Limits Report, Figure 2.4-1 (power-to-flow map). No core flow instabilities were observed and no reactor protection system challenges occurred.

The VY staff determined that the cause of the recirculation pump trip was personnel error. The error involved a contractor electrician conducting routine scheduled meter calibrations. While performing the calibration of the T-7-1A transformer local ammeter (B phase) using a generic work procedure, the electrician mistakenly left the associated ammeter circuit overcurrent trip relays (51B and 50B) un-bypassed. Upon inserting the calibration test current to the ammeter, the overcurrent relays tripped causing the 4160V to 480V station service transformer T-7 supply breaker to trip which de-energized Bus 7. The de-energizing of Bus 7 removed the "B" recirculation pump motor-generator set lubricating oil pump from service, causing the motor-generator set to trip on low lubricating oil pressure, and causing the "B" recirculation pump trip.

Following verification of the cause of Bus 7 being de-energized, the bus was re-energized at 11:55 a.m. and electrically loaded. The calibration of the T-7 transformer local ammeter was aborted and all subsequent scheduled meter calibrations suspended following a detailed root cause evaluation and the implementation of corrective actions. The "B" recirculation pump was returned to service at 1:56 p.m. and reactor power subsequently restored to 100 percent.

b. Observations and Findings

The inspector verified that reactor systems and balance of plant systems responded, as designed. One plant computer (ERFIS) software problem was identified involving the failure of the flow instability monitoring program (SOLOMON) to properly initiate upon B recirculation pump trip. This software programming anomaly was promptly and appropriately addressed prior to the conclusion of the inspection period.

The inspector observed control room operators actions to exit the Buffer Region of the power-to-flow map. This potential flow instability region was exited by 12:17 p.m. and the inspector noted prompt and appropriate actions by the control room operators to exit this operating region. The shift supervisor demonstrated effective control and coordination of the shift crew

and the supporting plant staff who responded to this event. The shift crew communications were clear and concise with an excellent team approach to addressing this operating challenge.

Operators did experience some minor difficulty in selecting control rods per the insertion sequence procedure, but were successful in addressing this rod select problem. The problem encountered was subsequently traced by the instrumentation and controls (I&C) staff to high resistance switch contacts in the reactor manual control system rod select matrix. The effected pushbutton selector switch was replaced and successfully post-maintenance tested. The inspector reviewed the I&C staff actions to address this operating concern and considered them prompt and appropriate.

c. Conclusions

Overall plant staff response to this event was good. Coordination and communications were effective in promptly diagnosing the cause of the "B" recirculation pump trip and in stabilizing the reactor plant. The root cause evaluation for this event was not completed by the close of this inspection period and will be reviewed by the inspector for thoroughness and adequacy of corrective actions. This is an inspection follow-up item **(IFI 96-03-01)**.

2.2 Maintenance Procedures and Documentation

Single Rod Scram Time Testing Update (92901)

a. Background

As previously discussed in inspection report 50-271/95-25, the VY staff observed an increase in the individual control rod notch 46 drop-out times. This increase in scram time was attributed to an apparent degradation of the scram solenoid pilot valve (SSPV) VITON elastomer diaphragms and the SSPV endcap design. Since the VY staff discovered their SSPV diaphragm concerns in early November 1995, other licensees have experienced similar problems and the Boiling Water Reactor Owners' Group (BWROG) Regulatory Response Group (RRG) has developed interim recommendations regarding the VITON diaphragm issue.

By letter dated February 16, 1996, the BWROG RRG promulgated their interim recommendations to all affected boiling water reactor plants with dual-type SSPVs containing VITON diaphragms. The VY staff documented their endorsement and proposed implementation plans for these recommendations by letter to the NRC, dated March 25, 1996. This letter stated that VY intends to meet or exceed the RRG's recommendations for testing both the SSPVs and the alternate rod insertion (ARI) system.

As stated in their March 25 letter to the NRC, VY will test 15 control rods during each rod pattern exchange. Prior to the issuance of this letter, VY had single rod scrammed 5 control rods on January 9 and 15 rods on February 27 during scheduled rod pattern exchanges. An average of 0.005 seconds increase in notch 46 drop-out time was observed on February 27 for the 15 control rods tested. The inspector noted that this increase was slightly less than anticipated by the reactor engineering staff (responsible for monitoring

control rod scram time results). Based upon the RRG recommendation, VY has scheduled rod pattern exchanges and individual rod scram time testing for April 23 (56 day interval), June 11 (49 day interval), and August 23 (73 day interval). In addition, VY stated that the recommendations for ARI system testing were verified to have already been instituted via the existing surveillance testing procedures.

b. Observations and Findings

The inspector verified that the number of control rods selected for scram time testing and the interval between testing was consistent with the RRG recommendations. As previously discussed in inspection report 95-25, the VY staff has demonstrated and continues to demonstrate an aggressive approach to understanding and monitoring the recent performance of the control rod SSPVs. This aggressive approach is again reflected in their commitment to the BWROG RRG recommendations. Notwithstanding, VY TS 4.3.C.2 states that scram time testing of 50 percent of the control rod drives in each quadrant be conducted "not more frequently than 16 weeks nor less frequently than 32 weeks intervals." The inspector notes that the lower bound (16-week interval) to this scram time testing requirement limits the frequency of testing of certain control rods. The control rod scram time testing schedule outlined in the March 25, 1996 letter potentially conflicts with TS 4.3.C.2, if appropriate controls are not in place to ensure the proper rod selection for this testing. Pending VY staff review of this observation and inspector verification of the procedural controls in place to ensure proper control rod selection and TS compliance, this issue is unresolved **(URI 96-03-02)**.

c. Conclusion

The VY staff's approach to monitoring and understanding the scram solenoid pilot valve VITON diaphragm degradation issue has been and continues to be aggressive. However, VY's increased frequency of individual rod scram time testing may potentially conflict with Technical Specifications 4.3.C.2 if appropriate administrative controls are not instituted. As discussed above, this issue is unresolved.

2.3 Quality Assurance in Maintenance Activities

LCO Maintenance Postponed (62703, 40500)

a. Observations and Findings

During this inspection period, VY station management postponed scheduled safety system limiting condition for operation (LCO) maintenance outages on two separate occasions. The first instance involved a planned maintenance outage for the reactor core isolation cooling (RCIC) system scheduled to commence the week of March 3. The inspector observed frequent discussions during the preceding weeks' morning meetings concerning the readiness of the LCO maintenance plan and related engineering concerns involving seismic qualification and containment integrity. On Friday March 1, the Plant Manager postponed this maintenance activity based upon insufficient documented

resolution of a number of these concerns.

The second instance involved the postponement of the planned "B" emergency diesel generator (EDG) LCO maintenance outage scheduled to commence the week of March 17. A recommendation to the Plant Manager to postpone this EDG maintenance outage came from the PORC. On March 15, the PORC was reviewing the station battery room issue (reference Section 3.1.1) and concluded that it was not prudent to proceed with the "B" EDG maintenance outage due to the battery room block wall seismic qualification concerns. The PORC concluded that the potential increased risk to safety system emergency power supplies was proper justification to postpone this elective maintenance activity.

b. Conclusion

The inspector concluded that the VY staff proceeded cautiously and thoughtfully in assessing the prudence of proceeding with the above planned LCO maintenance outages. In both instances, VY chose to conservatively postpone the elective maintenance until a more appropriate system work window was available.

3.0 ENGINEERING

3.1 Conduct of Engineering

Battery Room Block Wall Seismic Qualification (93702, 37551)

a. Background

On March 12 the design engineering staff initiated Event Report (ER) No. 96-1066 to identify two errors made in 1982 in the seismic qualification calculations for the battery room masonry block wall separating the two safety related station batteries. The two errors involved the incorrect assumption that the wall was constructed entirely of solid concrete blocks (only the upper one-third of the wall is solid block and the lower two-thirds is hollow block), and that this common block wall was subjected to the static and dynamic loading of two, not one, battery racks (the battery racks on either side of the wall are seismically braced by the block wall via through-wall threaded bolts and metal brackets).

The corrected seismic qualification calculation, which is derived from a linear/elastic methodology, identified the battery room block wall to have exceeded the acceptance criteria (maximum allowable tensile stress) by approximately a factor of six. Consequently, the block wall was determined to not be in accordance with the plant design basis and an operability assessment of this nonconforming condition was initiated by the engineering staff. Using a seismic qualification methodology referred to as "arching-action" (not a methodology reviewed and approved by the NRC staff for VY applications), the VY engineering staff was able to analytically demonstrate that the existing block wall and attached battery racks would survive a design basis earthquake. Based upon this new engineering analysis, VY concluded the block wall could sustain a design basis seismic event and thus not adversely impact the operability of the station batteries.

The inspector observed that VY's evaluation of this nonconforming masonry block wall configuration and its potential impact on station battery operability were generally consistent with the guidance of Generic Letter 91-18. VY made an Emergency Notification System (ENS) call on March 18 in accordance with 10 CFR 50.72 (Event No. 30127) informing the NRC staff that the nonconforming condition placed the plant outside of its design basis. On March 28, VY representatives met with the NRC staff to discuss the details of the "arching-action" methodology and its specific application to the battery room block wall. Pending the NRC staff's final evaluation of the application of this unapproved methodology (for VY), this issue is unresolved **(URI 96-03-03)**.

b. Observations and Findings

In monitoring the VY staff's handling of this design basis issue, the inspector made the following observations:

- Initiation of ER 96-1066 on March 12 (and subsequent March 18 ENS call) was not timely based upon the determination on March 6 that the seismic qualification calculation of record was both in error and the results exceeded the acceptance criteria.

- PORC review of the station batteries' operability determination on March 15 was completed and consistent with Generic Letter 91-18 guidance. No written explanation or summary was provided in advance or during the meeting to the PORC members, but the PORC member discussion was thorough and with clear safety focus.

- The PORC decision to postpone (due to the station battery operability issues) the planned "B" EDG LCO maintenance outage (scheduled to commence March 17) was prudent and indicative of a good safety perspective.

- The station was slow to formalize the Basis for Maintaining Operability (BMO) document which identifies the bases for the interim acceptability of the nonconforming masonry block wall and the station batteries' operability assessment, and defines the corrective action plan to resolve the current design basis conflict.

c. Conclusion

The identification of this nonconforming design issue was excellent and demonstrated a good questioning attitude on the part of the individual engineer who identified the problem. The engineering and operating staffs' handling of this design discrepancy with respect to promptly dispositioning the station batteries operability impact was not timely. PORC's review of the operability determination was comleted and their decision to postpone the pending EDG LCO maintenance outage was prudent and indicative of an excellent safety focus.

3.2 Quality Assurance in Engineering Activities

Torus Water Temperature Administrative Limit (37551)

a. Background

In November 1995, during a review of the Final Safety Analysis Report (FSAR) for periodic update, the VY staff identified that elements of TS Amendment 88, approved on June 6, 1985, were not completely incorporated in all appropriate FSAR sections. Amendment 88 involved an increase in initial (maximum allowable operating) torus water temperature from 90 to 100 degrees Fahrenheit (F). Closer examination of Amendment 88 and supporting documentation revealed that other safety analyses and emergency core cooling systems (ECCS) were potentially adversely impacted by this torus water temperature limit change. Three specific areas impacted were: the loss of coolant accident (LOCA) containment analysis; maximum (post-accident) projected torus water temperature; and available net positive suction head to the core spray and low pressure coolant injection pumps. A preliminary re-analysis by the VY engineering staffs of the LOCA containment response concluded that the plant would remain within its design basis assuming initial pool temperature is at or below 90 degrees F. Consequently, by Standing order No. 19, dated December 1, 1995, VY administratively imposed a more restrictive torus water temperature operating limit of 90 degrees F.

b. Observations and Findings

The inspector noted that continued engineering review and analysis of the above concerns lead VY to initiate the 10 CFR 50.72 notification (Event No. 30175) on March 26. A summary of the engineering staff actions were documented in a March 25, 1996 memorandum responding to Potential Adverse Condition Report No. 96-02. The inspector found the March 25 memorandum well written and concise. The scope of the torus water temperature issue was well defined and recommendations to resolve the outstanding potential safety issues were appropriate. The inspector summarized VY's recommendations below.

- Finalize, via formal calculations, the preliminary assessment restricting torus water temperature to 90 degrees F (target date - April 30, 1996).

- Conduct additional reviews to confirm that the use of containment over-pressure is allowable for calculating NPSH to ECCS pumps.

- Conduct formal analysis for maximum torus water temperature with a spectrum of LOCA break sizes, including a stuck open safety relief valve. Utilize the new containment model for this analysis, when available, (target date - December, 1996).

- Utilize engineering design report (EDR) 94-05 to track FSAR updates of the various elements of TS Amendment 88.

c. Conclusion

The inspector concluded that the VY engineering and operating staffs have been timely and thorough in bringing this complex design basis issue to appropriate resolution, to date. The administrative restriction of maximum operating torus temperature to 90 degrees F, pending the final analysis and a potential TS amendment, was a conservative safety decision. Pending completion of the VY staff's formal analysis of this potential design basis conflict and NRC staff review, the resolution of this issue remains unresolved **(URI 96-03-04)**.

3.3 Miscellaneous Engineering Issues (92903, 92700)

3.3.1 (Closed) LER 50-271/96-01: Technical Specification 4.6.E Not Met Due to Components Not Included in the In-Service Test (IST) Program Scope, dated March 1, 1996

The valve testing discrepancies documented in LER 96-01 were identified as a result of VY's in-depth IST review initiated as a result of corrective actions for a Notice of Violation (reference inspection report 95-22 and LER 95-17). The inspector reviewed the IST deficiencies documented in LER 96-01, the BMO written to document the associated operability determinations and corrective action plans, and discussed these items with the responsible VY engineers. The inspector concluded that VY's operability determinations were adequately founded and that the interim and long term corrective actions to resolve the testing inadequacies were appropriate. The inspector did note that LER 96-01 did not discuss all of the IST valves addressed in BMO No. 95-07, Rev. 1, dated February 14, 1996. The valves not adequately tested per the IST program, as identified in BMO 95-07, and not documented in LER 96-01 were:

- High pressure coolant injection (HPCI) pump suction check valve (V23-32)
- HPCI discharge check valve (V23-18)
- Both standby liquid control (SLC) pump discharge check valves (V11-43A & V11-43B)

The VY staff reviewed the inspector's observation and likewise concluded that the above valves should have been reported. At the conclusion of the inspection, a supplement to LER 96-01 was being prepared to address these additional IST discrepancies.

The above stated licensee identified and corrected Inservice Testing Program violations are being treated as a non-cited violation, consistent with Section VII.B.1 of the "NRC Enforcement Policy." The inspector identified oversight in reporting all of the recently discovered deficiencies (in LER 95-01) was considered an isolated case.

3.3.2 (Closed) LER 50-271/96-04: Discrepancies Identified in the Appendix J Leak Rate Testing Program, dated March 1, 1996

The inspector determined that a self-assessment of the Appendix J testing program discovered these discrepancies. The self-assessment was initiated as

a corrective action to the broad engineering management issues identified as contributors to the Appendix R/Fire Protection Program problems identified in 1995 (reference inspection report 95-26 and Enforcement Action 95-268).

For each of the Appendix J testing discrepancies, VY performed an operability determination and developed corrective actions to resolve the issue. This information was documented in BMO No. 96-02. The inspector reviewed BMO 96-02 and LER 96-04 to assess the adequacy and thoroughness of VY's operability assessment and corrective action plans. The inspector concluded that the operability determinations were adequate and the corrective actions appropriate, as discussed in the following paragraphs.

The specific testing discrepancy involving the reactor building closed cooling water (RBCCW) system was dispositioned via structural engineers assessing the seismic qualification of the non-seismic Class 1 portions of the RBCCW system. Based upon followup discussions, the inspector considered this engineering assessment appropriate, but not well documented with respect to specific review criteria and methodology used, as well as, detailed RBCCW system walkdown observations. The testing discrepancy involving the core spray system valve was mitigated by the application of an alternate testing method. The inspector found this alternate testing method appropriate for an interim operability determination, but as identified by the VY staff, this alternate testing methodology requires an NRC approved exemption to the ASME Code for long term use. A Code exemption was being pursued per the corrective action plan.

Similarly, the specific Appendix J testing discrepancy involving the main steam isolation valves (MSIVs) was adequately being compensated for by an alternate leak rate testing methodology for an interim operability standpoint. However, VY plans to satisfy the requirements of their NRC approved exemption involving the MSIVs prior to the conclusion of the 1996 refuel outage. These licensee identified and corrected Appendix J testing violations are being treated as one non-cited violation, consistent with Section VII.B.1 of the "NRC Enforcement Policy."

4.0 PLANT SUPPORT

4.1 Radiological Protection Controls

Reactor Vessel Shield Block Removal Outside Design Basis (92904)

a. Background

On February 21, VY made a 50.72 notification (Event No. 30005) identifying that immediately prior to entering the refueling outages in 1990 and 1992 all three layers of concrete shield blocks were removed from above the reactor vessel while the reactor was operating at greater than zero percent power. These events were outside the plant design basis with respect to the 30-day radiation exposure to personnel in the Technical Support Center (TSC) following a design basis accident. Based upon conservative estimates, VY

concluded that the total dose to personnel in the TSC could potentially exceed five REM, which conflicts with the requirements of NUREG 0737, item II.B.2, and the VY design basis (reference LER 96-03).

b. Observations and Findings

The inspector determined that a procedural revision implemented prior to the 1995 refuel outage prevented more than one layer of shield blocks from being removed prior to that reactor shutdown and subsequent refueling. Follow-up by the inspector also identified that, in addition to radiation shielding, the reactor vessel shield blocks provide significant mechanical barrier protection of the primary containment against tornado or high wind generated missiles. Similar to the radiation shielding minimum thickness calculations developed by the VY staff, the minimum missile protection requirements (thickness of concrete) provided by the reactor vessel shield blocks is 18 inches of concrete (reference FSAR Section 5.2). Accordingly, one layer of shield blocks (24 inches in depth) provides sufficient missile protection under the postulated design basis event.

c. Conclusion

At the close of the inspection period, VY had not completed their root cause evaluation of this issue and had preliminarily attributed the cause to a lack of formal procedural guidance for the removal of vessel shield blocks. Pending completion of the licensee's root cause evaluation and NRC staff review, this issue is unresolved **(URI 96-03-05)**.

4.2 Status of Fire Protection Facilities and Equipment

4.2.1 Fire Dampers Not Properly Installed (92904)

a. Background

On March 13, VY made a 50.72 notification (Event No. 30104) identifying that 11 fire dampers installed in the ventilation duct work of the control room, cable vault, switchgear rooms, turbine lube oil room, and high pressure coolant injection room were found to be installed contrary to the vendor's installation instructions. These dampers were all installed subsequent to original construction and in response to fire protection program enhancements and 10 CFR 50, Appendix R upgrades made in the late 1970's and early 1980's. The inspector determined that all of the specific installation deficiencies potentially impact the damper closure capability due to thermal expansion during a fire. The installation deficiencies include: improper caulking material; damper sleeves improperly installed or not installed around the damper; and, dampers installed in metal ventilation duct work which is of insufficient gage to resist collapse during a fire. Two fire dampers were identified which may not close under design conditions, in that, they are gravity drop vice spring assist and the ventilation flow rate may potentially inhibit damper closure.

b. Conclusion

The inspector verified that compensatory fire watches were properly implemented and that the VY engineering staff was actively pursuing resolution of the identified damper problems. The inspector noted that the discovery of these damper design/installation concerns was the result of a broad programmatic review by VY of each element of their Fire Protection and Appendix R Programs. This programmatic review is part of an ongoing corrective action to the previously identified Appendix R Program violations (reference inspection report 95-26 and escalated Enforcement Action No. 95-268).

4.2.2 Switchgear Rooms Carbon Dioxide Suppression System Declared Inoperable (92904)

a. Background

On March 20, VY made a 10 CFR 50.72 notification (Event No. 30146) identifying that the East and West switchgear rooms carbon dioxide (CO2) suppression systems have been declared inoperable due to insufficient post-modification test data. The inspector determined that another aspect of VY's broad programmatic review of their Fire Protection Program (see Section 4.2.1) was a detailed review of the plant CO2 suppression systems. This review identified that no full discharge testing results could be found to support post-modification testing of the East and West switchgear CO2 suppression systems. These CO2 suppression systems were modified via a 1982 plant design change (PDCR No. 82-14).

The inspector verified that a compensatory hourly fire watch was established in accordance with Technical Specification 3.13.D, and that the CO2 suppression system remained functional. Follow-up by the inspector identified that the VY staff has tentative plans to conduct switchgear room differential pressure and test gas functional testing to demonstrate the CO2 suppression system's capacity to achieve and maintain a 50% CO2 concentration in the upper portions of the rooms. The date of this testing had not been established by the conclusion of the inspection period.

b. Conclusion

The inspector concluded that VY's immediate and planned corrective actions were appropriate. Based upon insufficient testing data, to date, the operability of the East and West Switchgear Room CO2 suppression systems is, and has been indeterminate. This issue is unresolved pending completion of testing and NRC inspector review of the test results **(URI 96-03-06)**.

5.0 REVIEW OF UFSAR COMMITMENTS

A recent discovery of a licensee operating their facility in a manner contrary to the Updated Final Safety Analysis Report (UFSAR) description highlighted the need for a focused review that compares the plant practices, procedures, and/or parameters to the UFSAR descriptions. While performing the inspections discussed in this report, the inspector reviewed the applicable portions of

the UFSAR the related to the areas inspected. The inspectors verified that the UFSAR wording was consistent with the observed plant practices, procedures, and/or parameters.

As discussed in Section 3.2.1 above, the VY staff identified discrepancies (specifically Figure 14.6-7) where the applicable UFSAR sections had not been revised subsequent to the NRC staff's issuance of TS Amendment 88 in June 1985. Also, as discussed in Section 4.1, the VY staff had on two occasions operated the unit with all of the reactor vessel shield blocks removed and thereby degraded the available primary containment missile protection. Both of these UFSAR issues are being addressed via the assigned unresolved item.

6.0 MANAGEMENT MEETINGS

6.1 Exit Meeting Summary

The inspectors presented the inspection results to members of VY management periodically throughout the inspection period and at the conclusion of the inspection on April 12, 1996. The licensee acknowledged the findings presented.

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

INSPECTION PROCEDURES USED

IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 62703:	Maintenance Observations
IP 61726:	Surveillance Observations
IP 71707:	Plant Operations
IP 92700:	Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 93702:	Prompt Onsite Response to Events at Operating Power Reactors
IP 37551:	Onsite Engineering
IP 71750:	Plant Support Activities
IP 92901:	Followup - Operations
IP 92902:	Followup - Maintenance
IP 92903:	Followup - Engineering
IP 92904:	Followup - Plant Support
IP 90713:	Review of Periodic & Special Reports

ITEMS OPENED, CLOSED, AND DISCUSSED

Open

96003-01	IFI	Recirculation pump trip due to maintenance personnel error.
96003-02	URI	Adequacy of administrative controls for control rod scram time testing per 4.3.C.2.
96003-03	URI	Battery room block wall seismic qualification methodology acceptability and operability determination impact.

96003-04	URI	Torus water temperature administrative limit reduced to 90 degrees from TS limit of 100 degrees.
96003-05	URI	Reactor vessel shield block removal outside design basis.
96003-06	URI	Operability verification testing of the East and West Switchgear Room CO2 Suppression System not performed.
96003-07	URI	Resolve inconsistencies in license change request documentation involving recombiner catalyst service lifetime and any potential impact on catalyst maintenance.

Closed

96001	LER	TS 4.6.E not met due to components not included in IST program scope
96004	LER	discrepancies identified in Appendix J leak rate testing program
96006	LER	potentially inoperable RHR SW valves