U.S. NUCLEAR REGULATORY COMMISSION

Region I

Report No.: 50-271/89-07

Docket No.: 50-271 License No.: DPR-28

Licensee: Vermont Yankee Nuclear Power Corporation
RD 5, Box 169
Brattleboro, Vermont 05301

Facility: Vermont Yankee Nuclear Power Station

Inspection At: Vernon, Vermont

Inspection Conducted: April 18 - May 30, 1989

Inspectors: Geoffrey E. Grant, Senior Resident Inspector
Michael M. Kohl, Acting Resident Inspector

Approved by: Donald R. Haverkamp 6/14/89
Donald R. Haverkamp, Chief Date
Reactor Projects Section No. 3C

Inspection Summary: Inspection on April 18 - May 30, 1989
(Report No. 50-271/89-07)

Areas Inspected: Routine inspection on daytime and backshifts by two resident inspectors of: actions on previous inspection findings; operational safety; security; plant operations; maintenance and surveillance; engineering support; radiological controls; licensee event reports; and, periodic reports.

Results:

1. General Conclusions on Adequacy, Strength or Weakness in Licensee Programs

The licensee past program for determining the continuing operability of check valves RHRSW-43A/B was inadequate. Although the forward-flow capability of the valves was periodically confirmed, the reverse-flow operability of these check valves was never determined. Reverse-flow stoppage is necessary to support the alternate cooling mode of operation during the low probability event of loss of offsite power coincident with loss of the Vernon dam. The licensee apparently did not recognize the importance of assuring the reverse-flow stoppage capability of these valves. Licensee subsequent actions to restore valve operability and develop a long-term surveillance plan were appropriate and adequately addressed the deficiency. (See Section 7.1)

A partial engineered safety feature actuation resulted from an inadequate procedure revision and incomplete engineering review of a modification. Although the licensee considered this an anomalous event unique to this set of circumstances, the inspector noted that deficiencies with root causes similar to this event have occurred as documented in IR 89-05, IR 88-03 and IR 88-14. Although not indicative of a trend, these events warrant continuing licensee review and emphasis on the correct execution of all phases of the complicated modification process. (See Section 9.1)

Licensee response to a contamination incident was carefully considered and well executed. Prompt action limited the potential consequences of the situation. (See Section 8.2)

2. Violations

One licensee identified violation was identified involving failure to maintain control of locked high radiation doors. Prompt and effective corrective action was taken. No notice of violation was issued. (See Section 8.1)

3. Unresolved Items

No new unresolved issues were identified during this inspection period.

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*The NRC Inspection Manual inspection procedure (IP) that was used as inspection guidance is listed for each applicable report section.

DETAILS

1. Persons Contacted

   Interviews and discussions were conducted with members of the licensee staff and management during the report period to obtain information pertinent to the areas inspected. Inspection findings were discussed periodically with the management and supervisory personnel listed below.

   * Mr. R. Grippardi, Quality Assurance Supervisor
   * Mr. S. Jefferson, Assistant to Plant Superintendent
     Mr. J. Herron, Operations Supervisor
     Mr. R. Lopriore, Maintenance Supervisor
     Mr. R. Pagodin, Technical Services Superintendent
   * Mr. J. Pelletier, Plant Manager
   * Mr. R. Wanczyk, Operations Superintendent
     Mr. T. Watson, I & C Supervisor

   *Attendees at post-inspection exit meeting conducted on June 8, 1989. Additionally, Mr. W. Sherman, State of Vermont, attended the exit meeting.

2. Summary of Facility Activities

   Vermont Yankee Nuclear Power Station (VYNPS or the plant) continued full power operations during this report period. Throughout the period short term scheduled power reductions to 80-95% full power were conducted weekly to perform routine surveillances on control rod drives, main turbine and bypass valves. On May 23, power was reduced to 91% to support offsite maintenance of a power distribution line. On May 24, power was reduced to 95% for a short period to facilitate troubleshooting of feedwater flow oscillations. On May 26, power was reduced to 80% due to distribution system problems. Licensee notifications to the NRC were made in accordance with 10 CFR 50.72 for engineered safety feature actuations on May 1 and 24 (see Section 6.1) and on May 22, for low level contamination found offsite (see Section 8.2).

   The licensee announced the appointment of Mr. E. V. Lindamood as Radiation Protection Supervisor effective May 1, 1989.

3. Status of Previous Inspection Findings

   3.1 (Closed) Unresolved Item 86-10-07: Post-Maintenance Testing Following Repairs to Scram Solenoid Valves. This issue concerned control rod scram time test failures following maintenance performed on the ASCO air-operated scram pilot solenoid valves. The technical issue of incorrect rebuild kits was addressed by a licensee Part 21 report issued September 16, 1986, and was reviewed and closed in IR 86-22.

Remaining issues were the adequacy of licensee post-maintenance testing and the sequence to be used in future post-maintenance single rod scram testing. At the time of the occurrence (June 1986), the licensee post-maintenance testing program was insufficient to detect the component failures in the scram solenoid valves. Subsequent changes to the licensee program have corrected this deficiency. In addition to pre-job training, mock-up training and increased QC inspection, the licensee performs pre-operational functional testing of the solenoids. Additionally, scram testing is used to verify hydraulic control unit operability. The inspector observed licensee implementation of the revised post-maintenance program during the 1989 outage. The inspector also observed in-process controls and QC during scram valve and solenoid valve maintenance. No deficiencies were identified. The licensee also revised OP 4430, "Reactivity Anomalies" to require that all rods used in the shutdown margin test and in-sequence critical test are operable. The licensee response to this unresolved item was adequate to prevent recurrence of the deficiency. This item is closed.

3.2 (Closed) Unresolved Item 88-03-03: Lack of Aggressive Investigation and Followup of Reportability of Inoperable Service Water Radiation Monitor. This issue concerned a licensee initial failure to recognize the reportability of an inoperable service water effluent radiation monitor and subsequent programmatic disconnects that perpetuated the deficiency. The general area of event reportability has been identified as a licensee weakness and is currently being tracked by open item 88-08-05. The programmatic deficiency involved a disconnect between the AP 0028, "Operating Experience Review and Assessment/Commitment Tracking" (action items) system and the AP 0010, "Occurrence Reports" system. Previously, potential reportable occurrence (PRO) reports that were determined to be non-reportable but had some AP 0028 action assignment were not reassessed under AP 0010 to determine reportability upon development of new information. The licensee implemented an informal method to correct this disconnect. The method, which requires engineering support department (ESD) review of PROs after AP 0028 actions are complete, effectively addresses the programmatic deficiency. The current implementation is being formalized as an approved change to AP 0028. This item is closed.

3.3 (Closed) Unresolved Item 88-03-04: Revise Procedures or Request Technical Specification Amendment to Delete Testing Requirements for RHR-32 and 33. This issue concerned a licensee failure to update technical specifications (TS) after plant design change request (PDCR) 81-12 terminated and blank flanged the reactor vessel head spray (RVHS) subsystem. The RVHS containment isolation valves, RHR-32 and 33, were still listed in TS Table 4.7.2 as requiring once-per-operating cycle testing. The licensee discontinued this testing after implementation of PDCR 81-12. Containment integrity was maintained throughout this period by a blank flange on the terminated piping. The licensee submitted a proposed TS change on February 2, 1989, which deletes these valves from TS Table 4.7.2 thereby removing requirements to test these valves. Additionally, the licensee plans to permanently remove these valves and cap the containment penetration at a future date. During the 1989 refueling outage the licensee removed the remote manual operator switches for RHR-32 and 33 but retained the valve position indication lights on the control room panel. Additionally, as a conservative measure, the licensee performed the required surveillance testing on these valves prior to removal of the remote operator switches. This item is closed.

3.4 (Closed) Unresolved Item 86-10-05: Review YNSD Engineering Evaluation of Setpoint Drift and Instrument Loop Accuracies. This issue concerned analysis of the basis for determining instrument loop accuracies and subsequent setpoint changes of safety-related instrumentation. Yankee Nuclear Services Division (YNSD) completed and promulgated report YAEC1562, "Accuracy Report of Selected Class 1E Equipment Installed at Vermont Yankee Nuclear Power Station", in 1986 with a subsequent Revision 1 issued in October 1988. The report details the methodology used to identify instrument loop accuracies and provides a summary of instrument loop accuracy calculations. The inspector reviewed the report in general and several of the loop calculations. The methodology used in development of the loop error deviations appeared appropriate. No errors were identified in any of the calculations reviewed. However, the inspector noted that PRO 88-77 (October 1988) detailed an error in the development of the loop error value for the main steam line low pressure instrumentation. A non-statistical error to account for instrument head correction had been omitted during the development of the total loop error value. This item was reviewed by the inspector at the time of discovery and determined not to have safety significance due to conservatisms incorporated into the administrative limit (825 psig) versus the technical specification limit (greater than or equal to 800 psig). Actual values determined during surveillance testing were always greater than the TS limit. However, omission of the head correction from the determination of loop error represented a flaw in the execution of the instrument accuracy program. This was an isolated instance and review of the other loop error calculations found them satisfactory. The loop accuracy report is revised following outages in order to incorporate any procedural or instrumentation changes made at the plant. The program appears effective to ensure continuing accuracy of safety-related instrumentation setpoints. This item is closed.

3.5 (Closed) Unresolved Item 87-23-02: Development of a Formal Program for Cold Weather Operations. This issue concerned lack of a formal licensee program to control preparations for cold weather operations. The issue was precipitated by freezing and cracking of a condensate storage tank (CST) flush line and repeated freezing of reactor building ventilation damper instrument air lines. The licensee implemented OP 2196, "Preparations for Cold Weather Operations", on February 27, 1989. The procedure provides a checklist of cold weather preparation activities and assigns responsibilities for execution and review. The excessive delay in development of this procedure has previously been identified (IR 88-20). One action item remaining in this issue is resolution of PORC follow item (PFI) 88-84-02. This licensee item addresses a systematic review of the heat tracing system. Review of the final resolution of this aspect of freeze protection will be accomplished during routine inspection activities. Unresolved item 87-23-02 is closed.

3.6 (Closed) Unresolved Item 88-08-03: Improvement of Post-Trip Review Process. This issue concerned weaknesses identified in the licensee post-trip review process. Deficiencies included minimal post-trip information documentation and superficial post-trip report reviews. The licensee modified AP 0154, "Post Trip Review", in October 1988 to require shift supervisor development of an event reconstruction. As a final solution to the inspector's concern, this action was insufficient. However, the licensee continued to assess the adequacy of AP 0154 and developed additional enhancements that were incorporated in revision 4 dated May 5, 1989. These enhancements represent a substantial improvement in post-trip documentation and review. Execution of the revised AP 0154 has not been required but improvements appear to address the concern. This item is closed.

3.7 (Closed) Unresolved Item 88-19-02: Corrective Actions to Avoid Future Power Oscillations. This concern addressed completion of final licensee corrective actions to clarify procedures for prevention of power oscillations when operating in low flow regimes. The licensee had experienced small power oscillations on October 29, 1988, while performing single loop operations to support maintenance activities. The event is described in detail in IR 88-19. Several issues remained unresolved at the close of the IR 88-19 report period including final corrective actions to prevent future oscillations. Several interim positions were developed by licensee management to ensure routine plant operations would not result in power oscillations. These actions included immediate procedure changes, standing orders to operations crews and a Manager of Operations (MOO) directive. These measures were effective short term corrective actions. In November 1988 the licensee incorporated the guidance provided by the BWR Owners Group (BWROG) for interim stability corrective actions. In December 1988 procedure changes were implemented to formalize and incorporate the guidance. In late December 1988 the NRC issued Supplement 1 to Bulletin 88-07 which basically required licensees to implement the previously promulgated BWROG guidance. In a letter dated March 6, 1989, the licensee provided the required response to Bulletin 88-07 Supplement 1.

The inspector reviewed the final procedural implementation of measures to prevent the occurrence of power oscillations. Procedures reviewed included: OP 0102, "Power Operations"; OP 2110, "Reactor Recirculation System"; OT 3118, "Recirculation Pump Trip"; OT 3117, "Reactor Instability Procedure"; ON 3147, "Loss of RBCCW"; OP 2403, "Control Rod Sequence Exchange with the Reactor On-Line"; OP 2428, "Single Loop Operation"; and, OP 4424, "Control Rod Scram Testing/ Data Reduction". These procedures adequately implement the required actions to prohibit power operations in known instability regions, methods to be used to exit instability regions if entered, and actions to be taken if instability occurs. In some cases the licensee has imposed more restrictive measures than required by either the NRC Bulletin 88-07 or BWROG guidance. This item is closed.

3.8 (Closed) Violation 88-19-03: Failure to Adequately Post Contaminated Areas in Accordance with Plant Procedures. This item addressed licensee failure to post as well as inadequate posting of radiological control areas. The licensee response to this violation was provided by letter dated January 13, 1989, and adequately addressed the root cause and corrective actions. One corrective action requiring completion was revision of AP 0503, "Establishing and Posting Controlled Areas". The licensee revised and issued AP 0503 on February 3, 1989, meeting the date committed to in the violation response. The revised AP 0503 provides clearer direction for control and posting of radiological control areas. The inspector verified implementation of AP 0503 requirements throughout the 1989 refueling outage and noted no discrepancies. Licensee performance in this area will continue to be assessed during routine inspection activities. This item is closed.

3.9 (Update) Violation 89-04-01: Failure to Implement Compensatory Measures for Inoperable CO2 Systems. This issue involved the identification of a lack of a design basis demonstration for the CO2 systems in the cable vault and in the diesel fire pump day tank room. To be considered operable a test must have been performed to prove the CO2 systems can achieve the required CO2 concentration in the room. In the cable vault, this concentration must be maintained for ten minutes. The licensee had not performed the required tests on the CO2 systems for either the cable vault or the diesel fire pump day tank room, and technical specification required fire watches had not been implemented. The licensee established the required fire watch for the cable vault on April 21, 1989, and for the fire pump day tank room on April 24, 1989. On May 17, 1989, the licensee performed a discharge test of the CO2 system in the diesel fire pump day tank room. Results were satisfactory and the system declared operable. A continuous fire watch remains for the cable vault while the licensee completes final planning stages for resolving the inoperability of this CO2 system. A test similar to the one performed on the CO2 system in the diesel fire pump day tank room is planned for the CO2 system in the cable vault. The licensee expects to complete this test in the near future. This issue remains open pending resolution of the cable vault CO2 system deficiency.

4. Operational Safety

4.1 Plant Operations Review

The inspector observed plant operations during regular and backshift tours of the following areas:

Control Room	Cable Vault
Reactor Building	Fence Line (Protected Area)
Diesel Generator Rooms	Intake Structure
Vital Switchgear Room	Turbine Building

Control room instruments were observed for correlation between channels, proper functioning, and conformance with technical specifications. Alarm conditions in effect and alarms received in the control room were reviewed and discussed with the operators. Operator awareness and response to these conditions were reviewed. Operators were found cognizant of board and plant conditions. Control room and shift manning were compared with technical specification requirements. Posting and control of radiation, contaminated and high radiation areas were inspected. Use of and compliance with radiation work permits and use of required personnel monitoring devices were checked. Plant housekeeping controls were observed including cont. of flammable and other hazardous materials. During plant tours, logs and records were reviewed to ensure compliance with station procedures, to determine if entries were correctly made, and to verify correct communication of equipment status. These records included various operating logs, turnover sheets, tagout and jumper logs, and potential reportable occurrence reports. Inspections of the control room were performed on weekends and backshifts including April 20, 24, 27 and May 4, 18 and 25, 1989. "Deep backshift" inspections were conducted as follows:

Date	Time
April 20	9:00 p.m. - 11:30 p.m.
April 27	9:00 p.m. - 11:30 p.m.
May 4	9:00 p.m. - 11:30 p.m.
May 18	9:00 p.m. - 11:30 p.m.
May 21	10:00 a.m. - 6:45 p.m.
May 25	9:00 p.m. - 11:00 p.m.

Operators and shift supervisors were alert, attentive and responded appropriately to annunciators and plant conditions.

4.2 Safety System Review

The emergency diesel generators, reactor core isolation cooling, core spray, residual heat removal, service water, residual heat removal service water, safety related electrical, and high pressure coolant injection systems were reviewed to verify proper alignment and operational status in the standby mode. The review included verification that (i) accessible major flow path valves were correctly positioned: (ii) power supplies were energized, (iii) lubrication and component cooling was proper, and (iv) components were operable based on a visual inspection of equipment for leakage and general conditions. No violations or safety concerns were identified.

4.3 Feedwater Leak Detection System Status

The inspector reviewed the feedwater leakage detection system and the monthly performance summary provided by the licensee in accordance with VYNPC letter FVY 82-105. The licensee reported that, based on the leakage monitoring data for April 1989, there were no deviations in excess of 0.10 from the steady state value of normalized thermocouple readings. The inspector had no further questions in this area.

4.4 Inoperable Equipment

Actions taken by plant personnel during periods when equipment was inoperable were reviewed to verify: technical specification limits were met; alternate surveillance testing was completed satisfactorily; and, equipment return to service upon completion of repairs was proper. This review was completed for the following items:

Date Out	Date In	System
4/21		Cable vault CO2 system
4/24	4/24	Alternate cooling tower subsystem
4/24	5/16	Diesel fire pump day tank room CO2
4/25	4/26	"D" RHRSW pump
4/30	5/23	"C" SW pump
5/3	5/3	"A" EDG
5/8	5/8	"B" Standby Gas Treatment System
5/10	5/10	RCIC
5/23	5/23	"A" RHRSW pump
5/23	5/23	"A" Containment cooling subsystem
5/23	5/26	"B" SW pump

4.5 Review of Temporary Modifications

In response to NRC unresolved item 88-14-06, the licensee established a temporary modification program in early May 1989. The new program completely revised AP 0020 and incorporated the previous concepts of lifted leads and jumpers, and mechanical bypasses. The adequacy of the new program to address the concerns in 88-14-06 is still under review.

Temporary modifications were reviewed to verify that controls established by AP 0020 were met, no conflicts with technical specifications were created, safety evaluations were prepared in accordance with 10 CFR 50.59 if required, and requests were reviewed and approved prior to installation. Implementation of the requests was reviewed on a sampling basis. The following request was reviewed:

89-030 -- May 16 Transfer of the passive seal on the reactor building equipment airlock door from inner to outer

Additionally, several temporary modifications were closed out during the report period. These were reviewed for completeness and adequacy of system restoration.

4.6 Review of Switching & Tagging Operations

The switching and tagging log was reviewed and tagging activities were inspected to verify plant equipment was controlled in accordance with the requirements of AP 0140, Vermont Local Control Switching Rules. The following switching and tagging orders were reviewed:

89-1124	#1 Fan - West Cooling Tower
89-1200	"D" RHRSW pump
89-1222	"A" EDG
89-1260	Reactor building outer door
89-1286	Reactor building outer door
89-1288	"B" service water pump
89-1294	Standby liquid control tank heater
89-1296	TIP drives

4.7 Operational Safety Findings

Licensee administrative control of off-normal system configurations by the use of temporary modifications, and switching and tagging procedures, as reviewed in Sections 4.5 and 4.6, was in compliance with procedural instructions and was consistent with plant safety. Backshift inspections have consistently found operators to be alert and attentive. Operations are routinely conducted in a professional manner in an atmosphere of quiet control and competence. With the exception of isolated instances, overall plant cleanliness and material condition continue to be good. No deficiencies were identified in licensee operations associated with the reviews covered in Section 4.

5. Security

5.1 Observations of Physical Security

Selected aspects of plant physical security were reviewed during regular and backshift hours to verify that controls were in accordance with the security plan and approved procedures. This review included the following security measures: guard staffing; vital and protected area barrier integrity; maintenance of isolation zones, and, implementation of access controls, including authorization, badging, escorting, and searches. No inadequacies were identified.

6. Plant Operations

6.1 Primary Containment Isolation System Actuations

May 1 -- Group III

On May 1, 1989, with the reactor operating at full power, two primary containment isolation system (PCIS) Group III actuations occurred. The actuations, which isolate primary and secondary containment ventilation and initiate the standby gas treatment (SBGT) system, were caused by spurious activation of the "B" reactor building ventilation radiation monitor. At the time, the "A" channel monitor was reading normal. After verifying normal radiation levels and system operation, operators reset the PCIS actuations and returned the SBGT system to normal standby mode. System operability requirements were satisfied at all times.

The licensee determined that the spurious actuation was caused by a failed sensor/converter in the "B" reactor building ventilation radiation monitor. The failed unit was subsequently replaced and tested satisfactorily. The inspector noted that this failure was nearly identical to the failure and subsequent PCIS actuations of January 31 and February 13 (IR 89-02 Section 6.4). The licensee has not determined the root cause of the sensor converter failure. The equipment vendor is unaware of similar failures. The licensee is furnishing the failed unit to the vendor for further diagnosis. Licensee progress and actions for final resolution of the failure mode are appropriate and appear to be adequate. The inspector had no further questions in this area.

May 24 - Group III

On May 24, 1989, with the reactor operating at full power, a PCIS Group III and subsequent SBGT system actuation occurred. The actuation was inadvertently initiated while Instrument and Control (I&C) personnel were performing calibration procedure OP 4326, "Reactor Building Ventilation and Refueling Floor Radiation Monitors Functional/Calibration". The isolation was promptly reset and systems were restored to normal operation after plant personnel verified acceptable radiological conditions.

After completing checks on channel 452A, an I&C trainee under the direct supervision of a senior I&C technician repeated the procedure for channel 453A. The channel 453A trips were placed in bypass per procedure. The next step was to place the mode selector switch on the indicator trip unit for channel 453A to the zero and trip test positions. The trainee inadvertently operated the mode selector switch for channel 452A vice channel 453A. The mode selector switches for channels 452A and 453A are several inches apart. The trainee was under the direct supervision of the senior I&C technician throughout the procedure. However, due to the close proximity of the switches and the short period of time involved in repositioning the switch, the supervising technician was unable to prevent the error. The root cause of this event was human error.

The PCIS Group III and SBGTS operated as designed and successfully isolated the primary and secondary containment ventilation. A PCIS Group III isolation and subsequent SBGTS initiation are the expected result of the mode selector switch for an unbypassed channel being removed from the operate position. The licensee corrective actions included a discussion of the event with all personnel involved. The licensee will continue reviewing the event to determine what other corrective actions are warranted. The inspector observed that the procedure was adequate and not a contributor to the inadvertent actuation. Through interviews with I&C personnel and a review of OP 4326, the inspector determined human error was the root cause of this event. The inspector had no further questions on this event.

7. Maintenance/Surveillance

7.1 RHRSW Check Valve Failure

During the 1989 refueling outage, radiographic surveillance of residual heat removal service water (RHRSW) check valve 43A indicated that the valve disc fastening nut had apparently loosened. This surveillance was performed as part of a program developed by the licensee in response to NRC IEB 83-03, "Check Valve Failures in Raw Water Cooling Systems of Diesel Generators". This bulletin addressed concerns with the integrity of check valve internals and the potential for the internals to become separated or dislodged. Although the specific reporting requirements for this bulletin addressed only diesel generator cooling water check valves, licensees were encouraged to review maintenance and in-service testing (IST) programs in an effort to prevent "gross and multiple check valve failures that can defeat functions of systems important to safety". The bulletin identified forward and reverse flow testing or valve disassembly and inspection as acceptable methods of confirming check valve internal integrity. Other licensee proposed "equally effective methods of assuring integrity" were also allowed. It should also be noted that the bulletin emphasis was on valve internal mechanical failures that could prevent forward flow through the valve. Licensee response to IEB 83-03 was to identify RHRSW43A/B as valves of concern. The licensee determined that valve integrity could be monitored as part of the normal monthly diesel generator surveillance. Additionally, the valves were included in the IST program on a five year cycle (coincident with an outage). Testing was to be by radiographic examination. Both valves were examined and found satisfactory in 1983. Examination of RHRSW-43A during the 1989 outage was in accordance with the licensee IST program. Upon identifying a possible loose valve disc nut, the licensee disassembled RHRSW-43A. Inspection of the valve body internals revealed that, although the nut was intact, there was extensive corrosion product buildup on all valve internal surfaces. In addition, the valve disc was found to be stuck in approximately a 70% open position and would not stroke closed due to the corrosion. Check valve RHRSW-43A was then cleaned and refurbished and returned to service. Based on the results of the above inspection, the corresponding check valve on the parallel "B" loop of the system was also opened and inspected. This valve, RHRSW-43B, was disassembled and was found to be stuck in approximately a 50% open position. It was also refurbished and returned to service. Valve blockage in both cases was attributed to microbe induced internal corrosion from aerobic bacteria.

Diesel generator cooling water is normally provided by the service water system via branch piping from the residual heat removal service water supply headers. In each of these two branch lines a swing check valve (RHRSW-43A/B) functions to prevent backflow into the main header when diesel generator cooling water is provided by the RHRSW pumps during operation of the alternate cooling system. During normal service water system operation, these valves remain open. These valves are the only check valves directly in the branch piping which supplies cooling water to the diesel generator. Diesel generator operation in this mode would only be required if a loss of normal power occurred coincident with a failure of the Vernon dam.

Based on the observed valve internal condition, it was apparent that both of the subject check valves had been in the stuck open position for an extended time period. During this period, cooling to the diesel generators, station air compressors, and various ventilation coolers would have been degraded while in the alternate cooling mode. The licensee performed a review of plant system configurations and availability that would exist after the coincident loss of the Vernon dam and all off-site power assuming the check valve failures. Based on the review, it was apparent that reasonable compensatory actions could have been performed to provide adequate equipment cooling and decay heat removal capability to allow safe reactor shutdown. The licensee concluded that even in this unlikely scenario, the design intent of the alternate cooling mode could have been maintained in a degraded mode.

To ensure continued valve full stroke capability and internal integrity, the licensee revised the testing program for these valves. The check valves will be verified to close via disassembly and inspection during the 1990 refueling outage. Isolation and closure of these valves can only be accomplished during refueling since they also supply other equipment necessary for plant operation. Concurrent with the disassembly effort, methods to limit internal corrosion of the existing valves will be implemented. Following the 1990 refueling outage, valve closure will be verified by disassembly and inspection on an alternating basis during each subsequent refueling outage. The valve internal integrity inspection period (per IE Bulletin No. 83-03) will be revised from five year intervals to every refueling outage on an alternating basis. This inspection and verification will coincide with the valve closure examination referenced above. During the 1989 refueling outage, VYNPC performed a review of service water system check valve performance. This review confirmed that the full stroke capability of all these valves is satisfactorily verified by existing procedures.

Findings

Licensee response to this condition was good. Inspection of RHRSW-43B following problem identification in RHRSW-43A was appropriate. Review of the susceptibility of the entire service water system to this problem was appropriate and well executed.

Although the radiographic test method chosen by the licensee for RHRSW-43 A/B was adequate to address the specific concern of IEB 83-03, it was not adequate to determine full valve operability. At the time, the licensee did not recognize the safety function of these valves in the closed direction.

7.2 RHRSW Globe Valve Stem Failure

The stem for residual heat removal service water (RHRSW) valve 89A failed in August 1988. The stem had been in service since replacement in 1976 under EDCR 74-22. The EDCR 74-22 modification implemented a vendor recommended design change to minimize excessive valve vibration. Subsequent to the August 1988 repair, the licensee determined that RHRSW-89A was installed backwards in the system. The RHRSW system flow entered the valve on stem/disc side (top) of the valve seat vice from the bottom of the valve seat. At that time, the licensee attributed the stem failure to fatigue induced by long term flow perturbations caused by the valve installation error. The stem was replaced and the valve tested satisfactorily. The licensee began planning for the eventual reorientation of RHRSW-89A at a convenient time. The stem for RHRSW-89A failed again during system testing in April 1989 in preparation for startup from the refueling outage. After extensive licensee analysis, the valve was repaired and tested satisfactorily prior to plant startup.

The detailed licensee analysis following the April 1989 failure identified several contributing factors. Without assigning relative contribution weights, the licensee determined a combination of factors caused the rapid (August to April) failure of the stem as follows:

-- Stem material design and fabrication weaknesses. Independent laboratory analysis indicated that the manufacturer either used inadequate or no heat treatment following the forging process of the stem flange. This resulted in a less than optimal microstructure in an area of the stem that experienced high stresses.

-- Excessive thrusting of the disc into the seat due to a valve orientation error. The Limitorque operators for RHRSW-89A/B are set to deliver a nominal 50,000 lbs. of thrust in the closed seat. For RHRSW-89B (correctly installed), this thrust is nearly negated at the end of the valve stroke as the disc nears the seat by the force of system fluid flow from under the disc. For RHRSW-89A, this thrust is fully felt in the closed seat and is actually increased by the system fluid flow from above the disc. This overthrust caused excessive seating requiring additional load on the stem during subsequent valve opening.

-- Incompatibility of replacement parts. Due to a convoluted sequence of events concerning replacement parts for the original valve installation, replacement parts for the valve modification, original equipment vendor part numbering errors, valve parts supplied by an alternate vendor who corrected the errors made by the original vendor, and probable licensee procurement errors, incompatible parts were used in the August repair of the valve. Specifically, the stem nut and stem combination resulted in an excessive nut-to-stem clearance. This resulted in stem opening forces being concentrated on a smaller stem flange area.

Each of these factors was addressed in the April 1989 repair of RHRSW-89A as follows:

-- The new stem was manufactured by a machining vice forging process. Additionally, the licensee modified the stem flange area to be a radius vice a notch, thus reducing the stress concentration.

-- The Limitorque operator closing thrust for RHRSW-89A was reduced to approximately 10,000 lbs. thus reducing the overall seating force.

-- Other valve components were machined to accept the new stem and provide closer tolerances. Specifically, the stem nut-to-stem dimension was reduced to better distribute stem stresses during the opening stroke.

Additional licensee corrective actions include performance of ultrasonic evaluations of the stem on a periodic basis to detect any flaw initiation, alteration of the valve opening logic to allow less flow thrust on the valve disc, and ordering of correct replacement parts from the current vendor. The licensee is reviewing attributes of the stem failure for possible 10 CFR 21 reporting.

Findings

The licensee maintenance and engineering investigation and analysis following the April 1989 RHRSW-89A stem failure was excellent. The attention to detail and diversified approach to problem resolution yielded a solid root cause analysis and corrective action program. However, the excellence of this effort also illustrates the incompleteness of the analysis following the August 1988 stem failure.

Current corrective actions appear adequate to ensure at least near-term operability of RHRSW-89A. Licensee plans to replace valve internals with vendor-supplied parts only if other circumstances require replacement or ultrasonic evaluations indicate development of a stem flaw appear to be appropriate.

The licensee could have been more aggressive in pursuing reorientation of RHRSW-89A during the 1989 outage. Although this might not have prevented the April failure of the valve, it would have at least removed one of the failure contributors in addition to restoring the system to an as-designed configuration. The licensee is reviewing plans to correct the valve installation error during the 1990 outage.

7.3 Surveillance Observations

Radiation Monitoring System

On April 19, 1989, the inspector observed the performance of surveillance procedure OP 4384, "Area Radiation Monitoring System Functional/Operating Cycle Test". This is a monthly surveillance performed by the I&C department. At the conclusion of the pre-surveillance briefing held by I&C personnel, the inspector discussed the evolution with those individuals involved in the procedure. All personnel appeared knowledgeable and aware of surveillance requirements. All required administrative approvals were obtained prior to the initiation of the surveillance. Test instrumentation was within the current calibration cycle. Four of the twenty-six alarms were found to be outside the surveillance tolerance. The out-of-tolerance conditions were corrected and properly documented. The inspector reviewed the procedure and determined it was adequate. No deficiencies were identified.

Emergency Diesel Generators

On April 17, 1989, the weekly surveillance procedure, OP 4126, "Diesel Generators Surveillance", was performed by the auxiliary operator. It was noted that the "A" emergency diesel generator (EDG) air compressor started at 240 psig and secured at 250 psig vice the specified range of 225-230 psig air compressor start and 240-245 psig air compressor stop. Using a high accuracy pressure gage, I&C personnel determined the pressure switch was operating within the allowable tolerance. Performing OP 4126 again resulted in similar out-of-tolerance values. Subsequent troubleshooting determined the installed pressure gage on the air receiver used for the surveillance procedure was not accurate enough to use 5 psig as a tolerance range. The installed pressure gages on the air receiver are 0-600 psi gages with allowed two percent error. A maintenance request was initiated to replace the pressure gages on the diesel generator air compressors with 0-400 psi gages with allowed 0.25 percent error. At the completion of the troubleshooting, the inspector independently verified all equipment in the maintenance boundary was restored to a normal line-up. The inspector reviewed the quality control inspection report associated with this activity. The report required the evaluator to record the "as-found" and "as-left" setpoints for the pressure switch. In addition, the evaluator was required to verify that the components in the maintenance boundary matched nameplate data and all equipment associated with the work was within required calibration dates. The inspector also reviewed the maintenance request and procedure OP 5361, "Diesel Generators A & B Instrument Calibration", both of which were part of the work package for this job. The inspector determined all reviewed procedures were adequate. Further investigation revealed procedure OP 4126 had recently been changed, narrowing the range in which the air compressor must start and secure. The currently installed pressure gages on the air receiver are not accurate enough for the acceptance criteria of the revised procedure. This appears to be an instance of inadequate procedure revision follow-through. The inspector had no further questions in this area.

8. Radiological Controls

8.1 Locked High Radiation Area Access

On April 13, 1989, a licensee radiation protection technician performing a scheduled surveillance of locked high radiation area access doors found door #10 open to the main condenser bay. Immediately upon discovery of the condition, the door was closed and locked, and the plant health physicist notified. Licensee investigation revealed that an unidentified individual passed through the door and did not physically verify that the door had latched closed. Evidently, the individual relied on the automatic spring closer to latch the door. However, a temporary communications cable had accidentally dropped from overhead and blocked the door from reaching a fully closed position. The door remained open for approximately two hours. With the door open, there was no positive control of individual access to the posted high radiation area.

On April 17, 1989, maintenance personnel identified another open locked high radiation area door. The door was immediately closed and locked. Licensee investigation determined that a defective automatic spring closer allowed the door to remain open and personnel did not adequately ensure door closure. In this instance, the door remained unlocked for approximately ten minutes. The licensee performed a surveillance of all locked high radiation area doors and found no other problems. The temporary cable which caused jamming of the first door was removed and the licensee increased the door surveillance frequency. Licensee analysis determined that the primary cause of these events was personnel error in failure to assure door closure after passage. A defective door closure mechanism and improper temporary cable installation were contributing causes. Licensee corrective actions included surveillance of all doors and door closing mechanisms, direction to plant supervisors to discuss these events with department personnel, and revision of staff training programs.

Findings

Although the licensee has encountered past problems with control of locked high radiation area doors (see IR 88-18), the current events appear to be isolated instances of personnel error. Although uncontrolled access to high dose rate areas (2 R/hr) was possible for up to two hours, no personnel overexposures or unusual doses occurred. However, failure to maintain positive control over access to a posted high radiation area with dose rates exceeding 1 R/hr is a violation of technical specification 6.5.B. Because this condition was identified by the licensee, reported in LER 89-18, was not related to corrective actions for any previous violation, and was of a low severity level, no notice of violation will be issued for this licensee identified item (50-271/89-07-01). Licensee corrective actions were comprehensive and adequately addressed the issue. Consequently, this item is closed.

8.2 Low Level Offsite Contamination

On May 17, 1989, one member of an eight person crew that had been working on the refueling floor alarmed the PCM-1B whole body frisker. The personnel were performing activities associated with spent fuel pool reracking which had commenced on May 16. It was determined that the individual's socks were contaminated to approximately 200 counts per minute (cpm) over background. Due to the location of the contamination on the individual and the limited number of potentially contaminated work areas at the site, the licensee narrowed the source of the contamination to a dressing area on the level below the refueling floor and concluded that only the eight individuals were likely involved. The area had recently been decontaminated. Because the time the contamination originated in the dressing area was unknown, the licensee, as a precautionary measure, surveyed the temporary residences of the crew members and found four pairs of socks and one pair of shoes with low levels of contamination, approximately 150-600 ccpm. No other individuals were affected and no other off-site contamination was found. Follow-up whole body counts of all crew members were negative. The licensee informed the State of Vermont on May 18. The Commonwealth of Massachusetts (where two workers temporarily resided) was informed on May 19. A Commonwealth representative accompanied the licensee during a survey of the worker temporary residences.

The licensee postulated that crew members, while changing clothes in the dressing area, tended to concentrate low levels of contamination on their socks. Several surveys of the area prior to the incident showed no contamination. The licensee concluded that small amounts of contamination were leaching out of the floor paint in the affected area. Initially, the dressing area was secured and isolated. After covering the floor in the area and taking other appropriate corrective actions, the licensee reopened the dressing area. Measures for permanent resolution of the leaching contamination are under consideration. The inspector found the response and follow-up to this incident to be carefully considered and well executed. Timely identification of the potential contamination source limited the potential for further contamination incidents. The inspector had no further questions in this area.

9. Engineering Support

9.1 Partial ESF Actuation

On March 30, 1989, with the plant shutdown for a refueling outage, an automatic start of both core spray (CS) pumps and the "A" and "C" residual heat removal (RHR) pumps ("A" loop equipment) occurred while the primary containment was being pressurized for a scheduled containment integrated leak rate test (CILRT). The pump starts were the result of a high drywell pressure signal which had not been correctly inhibited prior to test commencement. This was a partial actuation that did not result in discharge to the reactor vessel.

The contacts which generated the pump start signal were added to the pump start circuitry by an engineering design change in 1987. The design change enhanced the operation of the automatic depressurization system (ADS) by providing automatic ADS initiation and auto start of core spray and low pressure coolant injection (LPCI) pumps on sustained LO-LO reactor water level. The ADS logic design change added two new time delay relays to achieve auto ADS initiation and pump starts on a sustained LO-LO reactor water level. These relays input into the ADS logic to simulate the presence of the high drywell pressure signal. The new time delay relays pick up the high drywell pressure sensor relays and pump start logic for RHR pumps "B" and "D". Due to the limited number of contacts available on the new time delay relays, contacts off the drywell pressure sensor relays were used in the pump start circuitry for RHR pumps "A" and "C", and the two CS pumps. This achieved the same function since the new ADS time delay relays pick up the high drywell pressure sensor relays. The presence of the ADS drywell pressure sensor relay contacts in the RHR "A" and "C" and CS pumps start circuitry provided an unbypassed high drywell pressure signal which caused the pump starts. The RHR "B" loop pumps were not affected by the pump start signal because the logic signal originates directly from the time delay relays which require a sustained low water level signal in order to be picked up.

The normal high drywell pressure signals were bypassed by procedure prior to the CILRT using installed test switches. However, the design change discussed above provided an ADS high drywell pressure signal directly to the pump start circuitry of the CS and "A" loop RHR pumps. This resulted in unanticipated pump starts when the primary containment was pressurized for the CILRT. Because the high drywell pressure logic for the RHR and CS systems was not satisfied (due to the bypass of the normal high drywell pressure signal), a full system isolation and actuation did not occur.

The licensee determined that the cause of this event was an inadequate procedure. The CILRT procedure did not include actions necessary to bypass the ADS drywell pressure input signals to the RHR and CS logic systems. The licensee attributed this procedural inadequacy to three factors: the CILRT procedure was not an "issued" procedure at the time the ADS design change was implemented in 1987 and therefore did not require review; the ADS design change effect on the RHR and CS logic systems was subtle; and, when the CILRT procedure was reinstated for the 1989 outage only design changes implemented during the outage were reviewed for impact on the procedure and test.

Findings

The inspector reviewed the assessment of the event and basically concurred with the licensee analysis and conclusions. However, the licensee attributing the procedure inadequacy to the fact that the procedure was not in an "issued" status is an implied indication of poor licensee administration of the procedure review program. If the procedure was not in an "issued" status from 1986 to 1989, then when it was reinitiated in 1989 it should have been treated as a new procedure and subjected to all of the required development and review controls. Additionally, a major contributor to the event appears to be lack of an adequate initial engineering review to assess the impact of the ADS design change on the affected logic circuits. The licensee considered this an anomalous event unique to this set of circumstances. Although uncommon, the inspector noted that deficiencies similar to this event have occurred (IR 89-05 Section 4.2, IR 88-03 Section 6.4, IR 88-14 Section 9.1). The inspector found the initial licensee long term corrective action of correcting the CILRT procedure to be incomplete. Subsequent discussions with the licensee indicated that other actions and reviews had been accomplished. These additional actions, delineated in revision 1 to LER 89-16, appear adequate. The inspector had no further questions in this area.

10. Licensee Event Reporting (LER)

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

10.1 LER 89-16

The LER 89-16, "Primary Containment Leak Rate Test Caused Inadvertent Core Spray and RHR Pump Start Due to Inadequate Procedure", addressed a spurious partial initiation of emergency core cooling system (ECCS) equipment as a result of performing a Type A containment test during the 1989 outage. Details of this event appear in Section 9.1 of this report. This was an excellent LER providing ample details and analysis. However, the inspector noted a lack of corrective actions identified in the LER. Further investigation and discussions with the licensee indicated that several corrective actions that were implemented were not documented in the LER. The licensee subsequently submitted revision 1 to LER 89-16 to amplify corrective actions. With this exception noted, the LER fulfilled the above criteria.

10.2 LER 89-17

The LER 89-17, "Service Water Check Valves Inoperable Due to Corrosion of Internal Parts", addresses the failure of two service water check valves due to bacterial induced corrosion products. Details of this event appear in Section 7.1 of this report. The description and analysis of the event were thorough. Corrective actions appeared sufficient to prevent recurrence. This well written LER fulfilled the above criteria and no deficiencies were identified.

10.3 LER 89-18

The LER 89-18, "Discovery of Open Locked High Radiation Area Doors", addresses two instances of normally locked access doors to high radiation areas being found open. Details of these events are further described in Section 8 of this report. This was a comprehensive LER which provided a good event analysis and adequate corrective actions. No deficiencies were identified and the LER fulfilled the above reporting criteria.

10.4 LER 89-19

The LER 89-19, "Inadvertent Primary Containment Isolation System Actuations Due to a Malfunction of the Reactor Building Ventilation Radiation Monitor Sensor/Converter", details two spurious PCIS Group III actuations with subsequent SBGT system starts. Further details are provided in Section 6.1 of this report. The LER accurately portrays the sequence of events. The analysis of the event is limited due to the indeterminate failure mode of the sensor/converter. The LER fulfilled the above criteria.

11. Review of Periodic and Special Reports

Upon receipt, the inspector reviewed periodic and special reports submitted pursuant to Technical Specifications. This review verified, as applicable: (1) that the reported information was valid and included the NRC-required data; (2) that test results and supporting information were consistent with design predictions and performance specification; and (3) that planned corrective actions were adequate for resolution of the problem. The inspector also ascertained whether any reported information should be classified as an abnormal occurrence. The following reports were reviewed:

-- Monthly Statistical Report for plant operations for the months of March and April 1989.

-- Annual Radiological Environmental Surveillance Report. The report summarizes the results of the VYNPC environmental surveillance program for 1988. The program is intended to provide early indication of any accumulation of any plant generated radioactive material in the environment; provide assurance that the environmental impact of the plant is within limits; verify the adequacy and functioning of effluent controls and monitoring systems; and, provide a standby monitoring capability to support rapid impact assessments. The report concluded that plant operations had no significant radiological impact on the environment.

-- Annual Operating Report for 1988. This report was submitted in accordance with 10 CFR 50.59 and describes facility changes for which NRC approval was not required. The inspector noted an improvement in the licensee safety evaluation summaries for each of the changes.

12. Management Meetings

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