U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Report No. 85-10

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

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

Facility Name: Vermont Yankee Nuclear Power Station

Inspection at: Vernon, Vermont

Inspection Conducted: March 5 - April 1, 1985

Inspector: William J Raymond 4/15/85
W. J. Raymond, Senior Resident Inspector date

Glenn W. Meyer 4/18/85
G. W. Meyer, Project Engineer date

Approved by: L. E. Tripp 4/23/85
L. E. Tripp, Chief, Reactor Projects date
Section 3A, Projects Branch 3

Inspection Summary:
Areas Inspected: Routine, unannounced inspection on day time and backshifts by resident and regional inspectors of: actions on previous inspection findings; plant power operations, including operating activities and records; plant physical security; surveillance testing; maintenance activities; training and implementation schedule for the new symptom orientated emergency procedures; followup on IE Bulletin 84-03; plant procedures for degraded grid protection; followup on receipt inspection program findings; and, implementation of changes to meet the requirements of the Radiological Environmental Technical Specifications. The inspection involved 94 inspection hours.

Results: No violations were identified in 10 areas inspected. Operational status reviews identified no conditions adverse to safe operation of the facility. Items identified in Inspection Report 85-11 as having an inadequate receipt inspection were evaluated and found acceptable as regards safe operation of the plant. The licensee should review the readiness of plant operators to implement the new emergency operating procedures in June, 1985 following completion of training.

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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 personnel listed below.

Mr. D. Reid, Operations Superintendent
Mr. J. Pelletier, Plant Manager

2. Status of Previous Inspection Findings

2.1 (Open) Follow Item 84-07-03: Laboratory QC Program. The establishment of supplemental QC measures for the facility radiological laboratory was documented in a February 1, 1985 memo from the plant Chemist to the Operations Superintendent. A QC program using control charts in accordance with Regulatory Guide 4.15 was implemented as of January, 1985 for the tritium counter, the well counter and the multi-channel analyzer. Other measuring equipment will be added to this list. Acceptance criteria for trending laboratory analyses using the charts was established at 2 sigma and 3 sigma, respectively, for the warning and control limits. Plant procedures are being revised to incorporate the control charts and other applicable guidelines from Regulatory Guide 4.15 will be adopted.

No inadequacies were identified. Licensee actions to date appear adequate to meet the commitment to the NRC. This item remains open pending completion of licensee actions to augment his program in accordance with the requirements of Regulatory Guide 4.15, and subsequent review of the licensee's radiological laboratory procedures by the NRC.

2.2 (Closed) Unresolved Item 84-26-02: Update LER. The licensee submitted Revision 2 to licensee event report 84-11 on March 8, 1985 to provide additional information regarding the cause and corrective actions for valves that failed the 1984 Appendix J Type C leak rate test. This item is closed.

2.3 (Open) Unresolved Item 84-23-02: Stores Shelf Life Control Program. The NRC letter dated November 20, 1984 transmitting Inspection Report 84-23 requested licensee management to respond, in writing, to staff concerns related to the failure of the Quality Assurance Program to disposition components and spare parts purchased prior to the initiation of the shelf life control program. No response from the licensee was received as of March 18, 1985. The inspector requested the corporate Senior Engineer - Operations to provide the status of licensee actions on this matter. The licensee reported on March 22, 1984 that a response to address the Inspection Report 84-23 shelf life control issues will be submitted by May 1, 1985.

The inspector expressed his concerns regarding the length of time the licensee has taken to address this issue. This item remains open pending receipt of the licensee's response and subsequent review by the NRC staff.

2.4 (Closed) Follow Item 82-15-01: Inclusion of sources and detectors into SNM inventory and shipment procedures. The licensee revised SNM control procedures to include sources and detectors. Specifically, the inspector reviewed procedure OP 0400, Special Nuclear Material Inventory and Accountability Procedure, Revision 18, which specified the SNM control measures applied to detectors and sources as listed items on inventory summary forms. This item is closed.

2.5 (Closed) Violation 83-09-01: Inaccurate reference in installation procedure. The inspector had found the installation procedure for torus modifications to specify one weld procedure while the actual welding was being performed to another, similar weld procedure. The licensee determined that the second weld procedure was technically acceptable prior to its use, but the installation procedure had not been changed to reflect this. The licensee's response letter FVY 83-73, dated July 13, 1983, stated that the installation procedures had been revised to also specify the second weld procedure and that personnel were reinstructed in the need to insure installation procedures are accurate. The inspector reviewed paragraph 7.4.4 of procedure SPN-70115-700, Revision 1, April 15, 1983 to verify that the procedure had been revised to accurately specify both acceptable weld procedures. This item is closed.

2.6 (Closed) Follow Item 83-03-02: Procedure change for backup means to supply fuel pool water from outside of the Reactor Building. A past licensee analysis of fuel pool water level under post-accident conditions showed that five to seven days would be available for developing a backup means to supply fuel pool water before any stored fuel would be uncovered. An inspector had requested that the licensee change the procedure to provide for supplying the water from outside the Reactor Building, and the licensee had agreed to review such a procedure change.

The inspector reviewed OP 3101, Loss of Fuel Pool Level, Revision 3, July 30, 1984, which specifies a backup means to supply fuel pool water when the Reactor Building is inaccessible. This item is closed.

3.0 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; random observations of the secondary alarm station; verification of physical barrier integrity in the protected and vital areas; verification that isolation zones were maintained; and implementation of access controls, including identification, authorization, badging, escorting, personnel and vehicle searches. The inspector reviewed the circumstances involved in the malfunction of security equipment at 2:06 P.M. on March 11, 1985 and the compensatory measures taken by the guard force during the interim period. Compensatory measures were acceptable. A telephone notification was made to the NRC Duty Officer in accordance with 10 CFR 50.73 at 4:15 P.M. No inadequacies were identified.

4.0 Shift Logs and Operating Records

Shift logs and operating records were reviewed to determine the status of the plant and changes in operational conditions since the last log review, and to verify that: (1) selected Technical Specification limits were met; (2) log entries involving abnormal conditions provided sufficient detail to communicate equipment status, correction, and restoration; (3) operating logs and surveillance sheets were properly completed and log book reviews were conducted by the staff; and, (4) Operating and Special Orders did not conflict with Technical Specification requirements.

The following plant logs and operating records were reviewed periodically during the period of March 5 - April 1, 1985:

- -- Shift Supervisor's Log
- -- Night Order Book
- -- Auxiliary Operator Log
- -- Control Room Operator Log
- -- Jumper/Lifted Lead Log
- -- Maintenance Request Log
- -- Shift Turnover Checklists
- -- Radiochemistry Analysis Log
- -- Core Performance Typer-Log

No unacceptable conditions were identified.

5.0 Inspection Tours

Plant tours were conducted routinely during the inspection period to observe activities in progress and verify compliance with regulatory and administrative requirements. Tours of accessible plant areas included the Control Room Building, Reactor Building, Diesel Rooms, Control Point Areas, the Intake Structure and the grounds within the Protected Area. Control room staffing was reviewed for conformance with the requirements of the technical specifications and AP 0036, Shift Staffing. Inspection reviews and findings completed during the tours were as described below.

5.1 Systems and equipment in all areas toured were observed for the existence of fluid leaks and abnormal piping vibrations. Pipe hangers and restraints installed on various piping systems were observed for proper installation and condition. No inadequacies were identified.

5.2 Plant housekeeping conditions, including general cleanliness and storage of materials to prevent fire hazards were observed in all areas toured for conformance with AP 0042, Plant Fire Prevention and AP 6024, Plant Housekeeping. No inadequacies were identified.

5.3 The inspector monitored the feedwater sparger leakage detection system data and reviewed the monthly summary of feedwater sparger performance provided by the licensee in accordance with his commitment to NRC:NRR made in letter FVY 82-105. The licensee reported that, based on the leakage monitoring data reduced as of February 28, 1985, there were (1) no deviations in excess of 0.10 from the steady state value of normalized thermocouple readings; and (2) no failures in the 16 thermocouples initially installed on the 4 feedwater nozzles. No unacceptable conditions were identified.

5.4 The status of the Residual Heat Removal (RHR), RHR Service Water, Standby Gas Treatment System, High Pressure Coolant Injection, Core Spray, Diesel Generator Fuel Oil and Starting Air, and Reactor Core Isolation Cooling (RCIC) systems was reviewed to verify that the systems were properly aligned and fully operational in the standby mode. The review included the following: (1) verification that each accessible major flow path valve was correctly positioned; (2) verification that power supplies and electrical breakers were properly aligned for active components; and, (3) visual inspection of major components for leakage, proper lubrication, cooling water supply, and general condition. A detailed review of the diesel generator fuel oil and starting air systems was completed to verify that plant procedures and drawings matched the as-built configuration. All of the above safety systems were found fully operable during the inspection period. The items below warranted further inspector followup.

5.4.1 The inspector walked down the two diesel generator starting air systems using the valve checkoff list of OP 2126 and the drawing, G-191160, Sheet 7, Revision 1. The inspector found the same error in the drawing for both starting air systems in that the alternate supply line, which bypasses the receivers, was shown capped between valve V72-74A(B) and the compressor. However, the inspector found the line was not capped and was connected to the compressor discharge between the relief valve and the receiver supply line. The inspector informed the licensee, and drawing corrective update 85-83 was initiated to correct the error.

5.4.2 During a walk down of the diesel generator fuel oil system, the inspector observed the 2 inch drain line on fuel oil storage tank TK-40-1A to be heavily rusted between the tank and the drain isolation valve. The inspector informed the licensee, and Maintenance Request 85-0568 was initiated. A licensee representative stated that initial plans included measurement of pipe wall thickness and, if acceptable pipe thickness exists, reapplication of paint.

This item is unresolved pending completion of the licensee's assessment of the drain line and subsequent review by the NRC (UNR 85-10-01).

5.5 Radiation controls established by the licensee, including radiological surveys, condition of access control barriers, and postings within the radiation controlled area were observed for conformance with the requirements of 10 CFR 20 and AP 0503. Radiation work permits (RWPs) were reviewed to verify conformance with procedure AP 0502. Work activities in progress were reviewed for conformance with RWP requirements. No inadequacies were identified.

5.6 Implementation of the following jumper (J/LL) and mechanical bypass (MBR) requests was reviewed to verify that controls established by AP 0020 were met;

no conflicts with the Technical Specifications were created; requests were properly approved prior to installation; and, installation and removal was in accordance with the requests: J/LL requests 84-187, 85-10, 85-13 through 85-18, and 85-20 through 85-22. No unacceptable conditions were identified.

5.7 Analysis results from samples of process liquids and gases were reviewed periodically during the inspection to verify conformance with regulatory requirements. The results of isotopic analyses of radwaste, reactor coolant, off-gas and stack samples recorded in shift logs and the Plant Daily Status Report were reviewed. Sample results for the standby liquid control tank on January 1, 1985 showed that the boron concentration was maintained within technical specification limits. No inadequacies were identified.

6.0 Operational Status Reviews

The operational status of standby emergency systems and equipment aligned to support routine plant operation was confirmed by direct review of control room instrumentation. Control room panels and operating logs were reviewed for indications of operational problems. Licensed personnel were interviewed regarding existing plant conditions, facility configuration and knowledge of recent changes to the plant and procedures, as applicable. Acknowledged alarms were reviewed with licensed personnel as to cause and corrective actions being taken, where applicable. Anomalous conditions were reviewed further.

Operational status reviews were performed to verify conformance with Technical Specification limiting conditions for operation and approved procedures. The following items were noted during inspector reviews of plant operational status.

6.1 Plant operators declared the 'B' uninterruptible power supply (UPS) inoperable at 8:10 A.M. on March 5, 1985 and started alternate surveillance testing per Technical Specification 4.5.A.4. The UPS was removed from service to allow replacement of cell #172 under MR 84-429, which was found cracked and leaking electrolyte during routine operational surveillance by the licensee.

UPS B is the normal supply for MCC-89B, which provides power for recirculation and RHR system valves, and RCIC steam supply valve, V13-15. MCC-89B was placed on its maintenance tie to MCC-8B while the UPS was out of service. Since valve V13-15 is required to close in response to a break in the RCIC steam line, the RCIC system was also declared inoperable and the downstream isolation valve, V13-16, was closed, while power for MCC-89B was transferred from UPS-B to MCC-8B. The maintenance tie to MCC-89B was established and the RCIC system was declared operable by 8:15 A.M. following satisfactory testing of V13-15. Plant operators made a telephone notification to the NRC Duty Officer at 8:55 A.M. to report the temporary outage of a single train system in accordance with 10 CFR 50.72.

The inspector reviewed the completion of alternate testing on the ECCS systems and both diesel generators during the period UPS B was out of service. UPS B was tested and declared operable at 3:10 P.M. on March 5, 1985, following replacement of the defective cell. No inadequacies were identified.

The UPS system uses Exide Series 'E' EC-11 cells and both banks were replaced during the 1984 outage. The licensee identified three cracked cells following installation of the B batteries during the outage and while installing the A system. All other A and B system batteries, including cell #172 were inspected at that time. No other problems were noted. The cracked cells were shipped to the vendor for examination and evaluation, who determined that the cells were most likely cracked when they were dropped during shipment and handling. Following discovery of the crack in cell #172, all cells were reinspected on March 5, 1985, and no further problems were noted.

The inspector reviewed the replacement of cell #172 and the licensee's evaluation of its condition prior to replacement. The licensee concluded that the UPS battery was not degraded and could have performed its intended function with cell #172 in the as found condition. No inadequacies were identified.

7.0 Surveillance Activities

The inspector reviewed portions of the surveillance tests listed below to verify that testing was performed in accordance with administrative requirements. The review included consideration of the following: procedures technically adequate; testing performed by qualified personnel; test data demonstrated conformance with Technical Specification requirements; test data anomalies appropriately resolved; surveillance schedules met; test results reviewed and approved by supervisory personnel; and, proper restoration of systems to service.

+ OPF 4125.04, Drywell and Torus Atmosphere Oxygen Surveillance, 3/18/85
+ OPF 4115.01, Primary Containment Surveillance, 3/11/85
+ OP 2132, Average Power Range Monitors, Revision 8, 8/2/84

No inadequacies were identified regarding testing performed under OP 4125 and OP 4115. The item discussed below was identified by the NRC Licensed Operator Examiner and referred to the inspector for resolution. The item was also discussed with the licensee during the Examiner's exit meeting on March 21, 1985.

7.1 OP 2132 is an Operations Department procedure that provides operator instructions on how to operate and test the average power range monitor (APRM) channels. Procedure Section A.2 provides instructions to align the APRMs for normal operations. Procedure Section A.1 provides instructions on the performance of an APRM front panel functional check, including a check of the LPRM input counting circuitry. It is unclear whether OP 2132 as written performs a proper test of the APRM counting circuitry, in that it appears to allow more inoperable LPRM inputs to an APRM channel than is otherwise allowed by Note 5 of Technical Specification 3.1.1. Further, the technical specifications are unclear whether the minimum number of LPRM inputs required for an APRM to be considered operable is 13 for all APRMs, or 13 for APRM channels B and E, and 9 for APRM channels A, C, D and F.

Performance of OP 2132 Section A.1 is an option for the operator and not a functional check of the APRMs required by the technical specifications. The technical specification functional test and calibration requirements are satisfied by

Instrumentation and Control procedures OP 4302 and 4308, which incorporate the same front panel checks provided in OP 2132. The inspector noted during discussions with the Operations Superintendent that Section A.1 is rarely performed by Operations personnel.

This item is unresolved pending further review of OP 2132 and the APRM operability requirements in the technical specifications (UNR 85-10-02).

8.0 Maintenance Activities

The maintenance request log was reviewed to determine the scope and nature of work done on safety related equipment. The review confirmed: the repair of safety related equipment received priority attention; Technical Specification limiting conditions for operation (LCOs) were met while components were out of service; and, performance of alternate safety related systems was not impaired.

Maintenance activity associated with the following was reviewed to verify (where applicable) procedure compliance and equipment return to service including operability testing.

+ MR 85-429, UPS 'B' Cell #172 Leakage
+ MR 85-403, Switchgear Room Fire Doors
+ MR 85-404, Control Room Dose Assessment Calculator
+ MR 85-183, Drywell Sample Valve 75-C1
+ MR 85-425, Drywell Sample Valve 75-C1
+ MR 85-471, Drywell Sample Valve 75-B2
+ MR 85-511, Drywell Sample Valve 75-C1

No inadequacies were identified. The following item warranted inspector followup.

8.1 During a routine quarterly test of the drywell sample valves at 3:00 A.M. on March 11, 1985, the closed indication for valve FSO 109 75 B2 was lost following stroke testing of the valve. Plant operators also noted that operation of valve B2 caused the indication for valve C1 to flicker. Maintenance request 471 was submitted, but the closed position indication returned after subsequent stroking of the valve and no further work was done.

During a routine weekly test of the sample valves at 3:50 A.M. on March 18, 1985, sample valve 75-C1 showed an intermediate indication following stroking and maintenance request 511 was submitted. Valve indication returned to normal following subsequent stroking of the valve. The inspector reviewed the status of all 8 sample valves at 9:00 A.M. on March 18, 1985 and observed each valve as it was stroked through a cycle. No problems were noted.

The inspector interviewed I&C personnel and reviewed maintenance records and the operator's log to identify other instances of sample valve failures. Maintenance request 183 was submitted on January 28, 1985 after valve 75-C1 showed anomalous indications following testing. Maintenance request 425 was submitted on March 4, 1985 when valve 75-C1 showed anomlous indications following testing. Based on operator observations, it appears that the anomalous indications went away following subsequent cycling of the valve.

Licensee personnel suspect the indication problems are due to slight (marginal) misadjustments of the reed switches associated with the valves. It is possible that a problem exists that causes the valve to bind mechanically. Further investigation has been deferred until the 1985 outage to allow complete disassembly and inspection of the valve position indication circuits and the valve internals, if necessary.

The eight sample valves, FSO 109 75 A-D1&2, are installed on four drywell sample lines, with two valves in series per line. Technical Specification Table 4.7.2b requires that at least one valve per sample line be operable and capable of being manually closed by the operator following an accident or sample line break. The inspector noted that the sample valves are stroked weekly per OP 4125.04 and quarterly per OP 4115.01. No other valve problems were noted during the remainder of the inspection period. There has been at least one operable valve per sample line.

This item is unresolved pending completion of licensee actions under the above referenced maintenance requests to investigate and repair as necessary sample valves 75-C1 and B2, and subsequent review by the inspector (UNR 85-10-03).

9.0 Symptom Orientated Emergency Operating Procedures

9.1 During discussions with licensee personnel on March 4, 1985, the Assistant Operations Supervisor, a senior reactor operator, expressed his concerns regarding the adequacy of the training being provided for the new symptom orientated emergency operating procedures (EOPS), as well as the appropriateness of the current schedule to implement the new procedures by June 1, 1985. The SRO felt that the new material was too different from previous procedures and training for operators to be adequately trained and ready to implement the EOPs by June. Formal training had begun on the procedures in January 1985.

The licensee is committed to implementing the procedures in accordance with this schedule by the NRC Confirmatory Order dated June 12, 1984, as amended by letter dated September 12, 1984. Other constraints identified by licensee management that support meeting the above schedule include the interrelationship and dependency of other TMI Emergency Response capability upgrades upon the EOP implementation schedule.

This matter was reviewed during discussions with Operations, Training and licensee management personnel during the week of March 18, 1985. NRC Region I management personnel participated in the discussions. The licensee's training for operators on the new emergency procedures consists of the following three segments: (i) initial classroom sessions; (ii) simulator training at the Dresden facility; and (iii) classroom refresher training just prior to the implementation date. The inspector determined that current concerns regarding readiness to implement the procedures arose from operators who had attended only the initial classroom lectures, which were a combination of training/verification sessions where the operators talked through the procedures and provided comments on their adequacy. Operators who had also completed the simulator training generally felt more confident and ready to use the procedures.

Based on the above, it appears that licensee management should make a special effort to assess the readiness of the operational staff to implement the new EOPs upon completion of the scheduled training. This assessment should solicit feedback from the operators on how comfortable they are with the procedures. The licensee should seek relief on the Order commitments from the NRC staff, as necessary. The licensee's assessment of the operational staff readiness to implement the new EOPs by June 1, 1985 will be reviewed on a subsequent inspection (UNR 85-10-04).

9.2 During discussions with licensee personnel on March 4, 1985, a potential concern was identified with the proposed symptom orientated EOPs apparent reversal of the heretofore established philosophy to assure adequate core cooling at all costs. The new EOPs provide instructions that would have the operator divert cooling water from the core to the containment as necessary to maintain containment integrity, even at the cost of loss of adequate core cooling. The inspector reviewed the licensee's proposed procedure OE 3103 for conformance with the BWR Owners Group guidelines, which have been accepted by the NRR staff.

Generic Letter 83-05 contains the staff's safety evaluation of the BWR Emergency Procedures Guidelines, Revision 2, as found in NEDO-24934 dated June, 1982. The SER provides a discussion (pages 17-20) of the primary containment pressure (PC/P) section of the primary containment control guideline. The PC/P section gives the rationale for actions to be taken if the Primary Containment Pressure Limit is exceeded.

As documented in the SER, the NRC staff concluded that it is acceptable to give preference to actions that will maintain containment integrity, even at the cost of temporary loss of core cooling, in order to assure that the final barrier to the release of radioactivity is preserved. This philosophy recognizes the importance of preserving the last release barrier for accidents involving severely degraded cores, and recognizes the dependence of the future success of the core cooling function on the integrity of the primary containment.

The version of OE 3103 reviewed by the inspector was consistent with the approved staff guidelines.

The licensee's implementation of the new EOPs in accordance with the requirements of NUREG 0737 Item I.C.1 will be examined further on a future NRC inspection. No inadequacies were identified.

10.0 <u>RETS Implementation</u>

The Radiological Environmental Technical Specifications (RETS) were issued with Amendment 83 to the facility license and became effective on April 1, 1985. New trip setpoints for the Stack Gas, Air Ejector, and Offgas process radiation monitors became effective with the RETS. The licensee completed setpoint changes for these monitors on April 1, 1985 in accordance with Setpoint Change Requests 85-13, 14, and 15. Additionally, the Air Ejector monitors (17-150A&B) were reclassified as non-safety related by the RETS and the associated automatic isolation function for offgas valves OG-516A&B was removed in accordance with Jumper and Lifted Lead Request 85-22 and Maintenance Request 85-0537. The alarm function associated with recorder 17-152 was also removed. Safety evaluations were completed as required for the above changes. The changes were completed in accordance with the administrative procedure for electrical jumpers.

Implementation of new requirements under the RETS will be examined further during future routine inspections. No inadequacies were identified.

11.0 IE Bulletin 84-03 Review

Licensee responses and actions taken for IE Bulletin 84-03, Refueling Cavity Water Seals, were reviewed to verify that: (i) the bulletin was received onsite, reviewed for applicability to the facility; and, (ii) corrective actions taken, or planned, were appropriate. Licensee actions on the item were as discussed below.

The inspector reviewed licensee letter FVY 84-138 dated November 26, 1984, which responded to the bulletin, and the Yankee Atomic Electric Company (YAEC) analysis, Evaluation of Potential Failures of the Spent Fuel Storage System, dated March 15, 1985, which provides the basis for the conclusions of the response letter. Also, the inspector reviewed the drawings for the inner and outer seals, the Reactor Building pool liner, and the fuel pool cooling and cleanup system.

In the above references, the licensee states that the refueling cavity water seals are flexible stainless steel bellows which are permanently welded in place and contain no pneumatic components. Based on their evaluation of the refueling cavity including the bellows, the licensee concluded the following:

1. The bellows are not subject to catastrophic failure. Both bellows have drains and alarms to detect any possible leakage. Also, the inner seal bellows contain a self-activated backup seal.

2. The design of the fuel pool and the reactor well prevent uncovering of the stored fuel, thus preventing any fuel damage.

3. In the event of any leak, more than adequate water makeup capacity is available from numerous sources, including up to 7500 gpm from the Residual Heat Removal System.

4. The maximum credible refueling cavity leak is 2385 gpm if the four reactor well drains simultaneously fail. Given such a leak and a failure to supply makeup water, the operators would have approximately 54 minutes to take action before the lower fuel pool level could

potentially uncover a fuel assembly in transit or in the fuel preparation machine. This time is acceptable, since under fuel assembly movement conditions, a licensed senior reactor operator would be required to be present in the refueling cavity area.

5. Existing plant procedures contain adequate instructions for actions to be taken in the event of loss of fuel pool and reactor well level.

6. The bellows are protected from the potential impact of a dropped fuel assembly by means of 1 inch steel plates above the bellows.

7. A 500 gpm leakage rate would occur if a 2½ inch diameter hole or a 20 inch long by ¼ inch wide split occurred in the bellows. These defect sizes are judged to be in excess of credible defects in the bellows. Based on evaluations above, such a leak would be well within makeup and recovery capabilities.

The inspector reviewed Emergency Procedure OP 3101, Loss of Fuel Pool Level, Revision 3, and Operating Procedure OP 2184, Fuel Pool Cooling System, Revision 10, to verify that appropriate guidance existed for recovery actions to seal failure and other causes of reactor well water loss.

The inspector concluded that the licensee's response to Bulletin 84-03 was acceptable.

The YAEC analysis of the bellows includes a recommendation that the bellows be inspected during the next refueling outage. The welded, enclosed installation of the bellows normally prevents any visual inspection. Therefore, special actions would be required to inspect them. The licensee is currently evaluating this recommendation. This item is open pending inspector review of the licensee's action on the recommended inspection of the bellows (UNR 85-10-05).

12.0 Quality Assurance Program Followup

12.1 Followup of Items with Inadequate Receipt Inspection

Discrepancies were identified in the licensee's receipt inspection program during an inspection by regional personnel on March 11-15, 1985, as described in Inspection Report 85-11. One issue resulting from the discrepancies concerned the potential for components and material that did not receive any receipt inspection to have been installed or used in safety related systems. The licensee was requested to review and evaluate the acceptability of material procured under the purchase orders identified in Attachment 1 to this report. For each order, the licensee was requested to identify the location of the material, and to provide the basis for his conclusion that the components were performing their intended functions if the material was installed in the plant.

The licensee's actions were summarized in a memorandum to the Administrative Supervisor dated March 27, 1985, as supplemented by Revision 1 dated March 29, 1985. The licensee determined which material was still in Stores, and which components were installed in the plant. An engineering evaluation was completed for material installed in the plant to address the significance or potential adverse impact of each component in service in a safety class system. The evaluations were based on periodic surveillance testing of the component or associated system, observations or demonstrations of system/component operability, post installation testing of the component or associated system, and other inspections performed prior to or during installation of the component. The evaluation was further supported by the results of supplemental inspections of a representative (at times 100%) sample of the remaining items in the purchase order, and by the satisfactory receipt inspection results of material from other line items in the purchase order.

The licensee concluded for the material described in Attachment 1 that there was assurance that components were performing their intended function and that there was no unreviewed safety question created by use of the material in the plant. A summary of the licensee's evaluation for each purchase order is provided in Attachment 1. The inspector reviewed the licensee's evaluation and identified no inadequacies. Further, the inspector confirmed by independent review and observation that operability has been adequately demonstrated for the plant emergency diesels and the reactor control rods.

The inspector had no further comments on this item at the present time. Resolution of the NRC concerns stemming from the deficiences in the receipt inspection program will be tracked through Inspection Report 85-11.

12.2 Peer Inspection Program Documentation

The licensee had reviewed and identified programmatic weaknesses in the receipt inspection program prior to the NRC inspection in this area. The licensee's findings and conclusions were summarized in a memorandum to the Manager of Operations dated February 22, 1985. Recommendations were made to effect improvements that when implemented would, in part, better document the actual inspections and reviews done during the receipt inspection process. The recommendations include actions to do more meaningful material inspections through improved/augmented component drawing files, tools and training of inspection personnel. The implementation of these recommendations will be included in followup to Inspection 85-11.

During a discussion on March 19, 1985, the Plant Manager summarized the results of a similar evaluation that had been completed in the area of 'peer inspections' for inplant maintenance and other activities. Based on Operational Quality Assurance (OQA) evaluations of the area, the licensee identified similar concerns regarding the documentation of the peer inspection process, and identified the need to reevaluate the entire program. Consideration is being given to supplementing peer inspection with an independent QC program. The initial plans were to begin the reevaluation and re-working of the peer inspection program in mid-March, 1985. The licensee decided to defer this effort pending resolution of the receipt inspection program issues. The OQA report for this item is scheduled to be issued in April, 1985.

This item is considered unresolved pending NRC review of the OQA report; and further review of the peer inspection process by the NRC staff, including the deficiencies identified by the licensee, the proposed resolutions and the implementation schedule for corrective actions (UNR 85-10-06).

13.0 Degraded Grid Procedures

By letter FVY 84-129 dated November 2, 1984 to NRC:NRR, the licensee proposed technical specifications that would establish operability requirements for Degraded Grid Protection System equipment installed during the 1984 refueling outage. Design changes were completed per EDCR 80-49 on August 20, 1984 in response to NRC initiatives to upgrade the protection provided for station 4KV electrical buses and thereby assure that potential low voltage conditions attendant with offsite electrical grid disturbances will not adversely affect safety related equipment. By letter FVY 84-46 dated May 15, 1984, the licensee described the control room alarm response procedures that would be implemented attendent with the design change to direct the proper operator response to a low grid voltage condition. The licensee stated that the alarm response procedure for annunciator 4-C on panel C-8 of CRP 9-8 would be rewritten, approved and implemented prior to startup from the 1984 refueling outage. The licensee's proposal is currently in NRC Region I for review and approval.

The inspector reviewed the status of licensee actions on this item. The EDCR 80-49 design changes were implemented during the 1984 outage, but the alarm response procedure was not upgraded as described in FVY 84-49. The new protective relays will cause the onsite electrical buses to automatically separate from the offsite supply if a low voltage condition occurs concurrent with an accident signal. The operator response to this situation would be no different under either the old or proposed procedures. Should a low voltage signal occur without an accident signal present, then operator response is required to assess the situation, take remedial actions if possible to restore voltage, or isolate the onsite buses from the offsite supply to protect plant equipment. There is a difference in the actions prescribed by the old and new procedures for this situation.

The licensee decided to not implement the new procedure due to technical problems with the proposed instructions, and due to other questions that arose (discussed below) that require resolution. During a discussion on March 13, 1985, the licensee stated that the Plant Operations Review Committee reviewed both procedures prior to restart from the outage and concluded that it was acceptable to operate under the old one since its instructions would direct more conservative operator actions than the proposed instructions, and adequate protection of station equipment would be assured.

During a telecon with the Operations Superintendent and the YAEC cognizant engineer for the design change on March 13, 1985, the inspector determined that details of the diesel load shed bypass and reinstatement circuitry are still under review. These questions include: (i) is additional hardware required to reinstate load shedding if the diesel breaker opens after the diesel has been on its bus; (ii) would ESF loads be sequenced onto the bus if the accident signal occurred sometime after the low grid voltage condition; and (iii) can the diesel carry both normal and ESF loads if load sequencing does not occur. Operating procedures would be redefined and hardware changes made as required upon completion of these reviews. The licensee is scheduled to complete this effort by May, 1985.

The inspector noted that a special inspection at the site is scheduled as part of the Regional licensing action review for this item. The above information and the status of the licensee's actions in this area were discussed with the cognizant NRC reviewers.

The inspector had no further comment on this item. No inadequacies were identified.

14.0 Management Meetings

Preliminary inspection findings were discussed with licensee management periodically during the inspection. A summary of findings for the report period was also discussed at the conclusion of the inspection and prior to report issuance.

ATTACHMENT 1

The following summarizes the licensee identified status of the specific items identified in Report 50-271/85-11 as having received inadequate receipt inspection.

1. PO No. 14409: 3 Valcor Engineering Corporation solenoid valves plus 2 kits to be used in the alternate control rod insertion system.

   Status: All items were in the stockroom and none were installed in the plant. Receipt inspection, properly performed, shows that these parts are acceptable for use.

2. PN No. 9706: To Fairbank Morse Engine Division for diesel generator parts.

   + PO Item 2 - 4 ASCO 3-way solenoid valves.

   Status: Four valves were ordered. Three remained in the stockroom. One valve was installed in the "A" diesel generator air starting system on October 12, 1983. The diesel was verified operable following installation of the new valve. The air start valve was tested for each quarter per OP 4126. The component has functioned satisfactorily during all tests performed since installation of the new valve.

   The other three air start valves purchased under the same purchase order have now been properly receipt inspected and found to be totally acceptable.

   + PO Item 3 - 4 Norgram Air Filters.

   Status: All items were in the stockroom and none were installed in the plant. These items are on QC hold pending completion of a proper receipt inspection.

   + PO Item 5 - (12) Air Filter elements. 4 non-transparent hermetically sealed boxes with 3 filters each.

   Status: All items were in the stockroom and none were installed in the plant. These items are on QC hold pending completion of a proper receipt inspection.

3. PO No. 22711: To Unistrut Corporation.

   + PO Item 3 - (200) ninety degree angles. 75 angles were issued and 125 were found in the stockroom.

   + PO Item 12 - (1500) 3/8" Unistrut spring nuts. 175 nuts were issued for plant related work. 1325 nuts were found in the stockroom.

   Status: The angles and spring nuts were used in many design changes during the 1984 outage for equipment environmental qualification upgrades. The material was used for the installation of conduit and there is no documentation that shows other inspections that may have been performed. An installation and test procedure was used to install the material, and personnel installing the material would have recognized any obvious defects.

The other nuts and angles from the same purchase order now have been inspected and found fully acceptable.

4. PO No. 22396: To Unistrut Corporation.

   + PO Item 3 - (100) ninety degree angles were ordered and 50 were issued for plant related work.

   Status: The same as that given for PO Item 12 above under 22711. The use of these angles in the plant has now been found to be acceptable based on receipt inspection showing acceptability of all of the 50 in stock and on the use of an installation and test procedure to install the parts used.

5. PO No. 22554: Hilti Kwik Bolt 3/8" x 2-3/4". 300 bolts were issued for plant-related work. 700 more are in stock.

   Status: The Hilti bolts were used in many design changes during the 1984 outage for equipment environmental qualification upgrades. Most of the material was used to install conduit and there is no documentation that shows other inspections that may have been performed. Bolt data forms were generated for the installation of other equipment and are available in the job order packages.

   An installation and test procedure was used to install the material, and personnel installing the material would have recognized any obvious defects. Installation personnel are knowledgeable of manufacturer installation instructions which provide specific requirements for the number of turn of nuts, thread engagement, and torque requirements.

   The other 700 bolts from the same purchase order have now been inspected and found fully acceptable.

6. PO No. 22706: Hilti Kwik Bolt 3/8" x 2-3/4". 200 bolts were issued for plant-related work. 1800 more are in stock.

   Status: Same as for PO No 22554 above. The use of the bolts in the plant was found to be acceptable based on installation controls and the recently determined acceptability of all 1800 of the in-stock bolts.

7. PO No. 13068: 125 Vac Coil Sep. Control (w). Zero-stocked code, yet item had a p-tag indicating it was safety-related.

   Status: The licensee determined that an administrative error had occurred while processing this item and it should not have had a p-tag affixed to it. This item is not safety-related.

8. PO No. 12259: Collet and piston. Two collet pistons were used in the plant.

   Status: The collet pistons were installed in control rod drives. One was installed in CRD S/N 7527 which was returned to the reactor at location 18-11 during the 1984 refueling outage. This rod subsequently could not be pulled beyond position 46. The location of the other collet piston is indeterminate, but it was concluded that it was installed in a CRD disassembled since 1979.

   Operational problems associated with a faulty collet piston would be exhibited by the inability of the CRD to latch at a specific notch, or the inability to unlatch from a notch once latched. This type of problem has not been observed on any CRD during the period from 1979 to the present. The rod at position 18-11 has been demonstrated to be fully operable and the problems associated with the rod are not attributable to the collet piston.

   Collet pistons were installed per OP 5211 by qualified GE personnel, who would recognize any defects that may have existed. Inspections performed per the requirements of OP 5211 far exceed those that would be performed under a receipt inspection. The other three identical collet pistons purchased under the same purchase order were receipt inspected and found fully acceptable.

9. PO No. 16480: Collet pistons. 6 ordered and none were issued for use in the plant.

   Status: All items were in the stockroom and none were installed in the plant. Receipt inspection now shows all 6 of these pistons to be fully acceptable.

10. PO No. 22041: Fuel Cartridge (DG). 3 fuel filters were issued for use in the plant.

    Status: The three fuel filters were installed on the "A" diesel generator; one on July 28, 1984 and two on February 13, 1985. There is no documentation of additional inspections that may have been performed. The filters were installed by experienced mechanics, who would have noted any obvious defects. The diesel was satisfactorily tested subsequent to installation of the fuel filters, which demonstrated operability of the filters. Filter delta-P measurements are taken during the monthly test and no unusual delta-P readings were noted.

    The other 15 filters purchased under the same purchase order have now been inspected and found totally acceptable.

11. PO No. 10269: Time Delay Relay. One item was ordered and was found in the stockroom.

    Status: This item was in the stockroom. Receipt inspection now shows it to be fully acceptable.

12. PO No. 18047: Agastat Time Delay Relay.

Status: The three relays associated with this purchase order were removed from stock in April 1983 and used in an attempt to rebuild an acceptable replacement for the "A" diesel stopping relay. All three relays and associated parts were scrapped and another relay was installed for an interim period. Two new model E-7024 PE 002 relays were subsequently purchased under PO 20824 and installed in both diesel generators. Acceptability of the relays used is indicated by post-installation functional testing and proper relay performance during regular diesel surveillance testing. However, adequacy of the parts purchased under PO 20824 is still under evaluation by the licensee.