U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-255/93008(DRP)

Docket No. 50-255 License No. DPR-20

Licensee: Consumers Power Company
212 West Michigan Avenue
Jackson, MI 49201

Facility Name: Palisades Nuclear Generating Plant

Inspection At: Palisades Site, Covert, MI

Inspection Conducted: March 23 through May 17, 1993

Inspectors: J. K. Heller, J. A. Gavula, A. H. Hsia, D. G. Passehl, D. W. Nelson, M. E. Parker, W. D. Shafer, J. F. Schapker

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DATE 5/28/93

Inspection Summary

Inspection from March 23 through May 17, 1993 (Report No. 50-255/93008(DRP))

Areas Inspected: Routine, unannounced inspection by inspection personnel from this office and the Office of Nuclear Reactor Regulation of actions on previously identified items, operational safety verification, maintenance, surveillance, engineering and technical support, reportable events, spent fuel handling machine malfunction, dry cask storage operations, loading of the first and second dry cask, and a quarterly management meeting. No Safety Issues Management System (SIMS) items were reviewed.

Results: One violation was identified involving failure to follow procedures during fuel moves (paragraph 8). One non-cited violation (NCV) was identified pertaining to diesel generator inoperability (paragraph 7d). One unresolved item was identified pertaining to an increasing trend in the number of operator errors (paragraph 3e).

The strengths, weaknesses, and inspection followup items are discussed in paragraph 1, "Management Interview."

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DETAILS

1. Management Interview (71707)

   The inspectors met with licensee representatives (denoted in paragraph 10) on May 24, 1993, to discuss the scope and findings of the inspection. In addition, the likely informational content of the Inspection Report with regard to documents or processes reviewed by the inspectors during the inspection was also discussed. The licensee did not identify any such documents or processes as proprietary.

   Highlights of the exit interview are discussed below:

   a. Strengths noted:

      1) The administrative programs for minimizing the risk to shutdown cooling during reduced primary coolant system inventory and the methods used to assure plant personnel were aware of the plant condition (paragraph 3d(3) and 5f).

      2) Management's involvement during activities that have the high potential to affect safety (paragraphs 3d(3), 9d, 10b, 10c(2), and 11.

      3) The prompt response to resolve defective anti-rotational keys in motor operated valves (paragraph 6a).

   b. Weaknesses noted:

      1) The licensee found the diesel generator hand switch for the fuel oil filter in an unspecified position (paragraph 3b).

      2) The licensee's difficulty in reducing the negative trend pertaining to operator errors (paragraphs 3e and 8).

      ?) A control room operator appeared to be overextended while performing control room activities (paragraph 10c(1)(c)).

   c. The violation for a failure to follow procedures (paragraph 8).

   d. The non-cited violation pertaining to diesel generator inoperability (paragraph 7d).

   e. The unresolved item pertaining to operator errors and the request to provide a docketed evaluation (paragraph 3e).

   f. The inspector discussed the potential for providing incorrect information to the NRC and potential consequences if an additional example is identified (paragraph 2d).

g. The two unusual events (two diesel generators inoperable and excess primary coolant leakage) were discussed (paragraphs 3b and 3c).

h. The licensee response to industry events that could affect the plant (paragraph 6).

i. Four recent reactor trips were the results of activities to improve plant performance. The need to proceed cautiously with future improvement modifications, and the potential consequences if additional trips were to occur, were discussed (paragraphs 7a and 7c).

j. The need to provide a brief narrative when using "N/A" to delete performance of a test (paragraph 10c(1)(b)).

k. The auxiliary building load distribution system design and calculation were reviewed. The inspector noted that problems with design calculations, such as occurred on the steam generator replacement project, did not recur (paragraph 9b).

2. <u>Actions on Previously Identified Items</u> (92701, 92702)

a. (Closed) Open Item 255/90018-02(DRP): Spent Fuel Pool (SFP) Leakage

The SFP bulk head gate developed a significant water leak in 1990 due to inadequate nitrogen pressure to the bulk head gate inflatable seals. This resulted in a SFP level reduction of approximately 6 feet when the water leaked to the transfer canal. The licensee's evaluation (D-PAL-90-193) documented that the leakage was caused by a faulty nitrogen pressure regulator. The licensee modified the SFP gate and the pressure regulating system. The modification included installation of a redundant nitrogen regulator and a bulk head gate with dual inflatable seals. There have been no additional leaks from the SFP bulk head gate.

b. (Closed) Open Item 255/90021-01(DRP) as updated in Inspection Report 255/90031(DRP): Safety Injection and Refueling Water Storage (SIRW) Tank Leakage

During the 1990 refueling outage a leak developed at several of the SIRW tank bottom plate "pipe to penetrations" welds. The licensee analyzed several failure mechanisms and concluded that the stresses during draining and filling cycles contributed to several cracked welds and the leak. The licensee repaired the cracked welds and implemented a surveillance program to confirm the continuing integrity of the welds. No additional leakage problems have been identified.

c. (Closed) Violation 255/91018-01(DRP): Safety related support equipment exceeded Limiting Condition for Operation (LCO).

PS-0918, "Component Cooling Water Pump Discharge Pressure Interlock Switch," was removed from service without consideration of Technical Specification applicability. PS-0918 was isolated as part of ground fault trouble shooting activities. The ground fault cleared when PS-0918 was isolated. PS-0918 remained isolated for several weeks while the ground fault was resolved. The safety significance was minor but indicated that the proper reviews were not performed after the equipment was removed from service. The failure to perform the reviews resulted in the violation.

The licensee's response dated January 19, 1991, to the Notice of Violation acknowledged that timely reviews were not performed after PS-0918 was removed from service. The corrective actions pertained to training of Corrective Action Review Board (CARB) members regarding timely operability determinations. The inspector routinely reviewed the operability determinations of the CARB and has not found additional examples of this violation.

d. (Closed) Unresolved Item 255/92027-1(DRP): Incorrect Information Provided to the NRC

The licensee provided incorrect information to the NRC on December 12, 1991, pertaining to proposed completion dates of modifications to safety related circuit breakers. This unresolved item required a docketed evaluation of the causes and corrective actions. The licensee's response dated February 16, 1993, documented that miscommunication between the licensing and engineering departments resulted in the submittal of incorrect information to the NRC.

The inspector independently developed a time line of the modification and confirmed that miscommunication between the licensee's work groups resulted in the submittal of incorrect information to the NRC. The inspector concluded that the delay in completing the modification did not create a safety hazard.

The licensee's corrective actions included staff training pertaining to the importance of NRC commitment dates. They also formalized the definition of an NRC commitment date. Additionally, a guideline for preparing submittals to the NRC was prepared.

The inspector evaluated this item for 10 CFR 50.9, "Completeness and Accuracy of Information," enforcement action and concluded that the incorrect information was not identified by the NRC or used by the NRC to schedule inspection activities. This meets the criteria specified in 10 CFR 2 Appendix C for not taking enforcement action. The inspector also determined that the licensee did not knowingly provide incorrect information to the NRC.

Additionally, on January 1, 1992, enforcement action was issued (Inspection Report 255/91026(DRS)) for several separate examples of when the licensee provided incorrect information to the NRC. The inspector concluded that if the incorrect information that resulted in this unresolved item was identified at the time the previous enforcement action was issued, then this would have been an additional example of that enforcement action. Additionally, if this item had been included with the enforcement action the severity level would not have changed. Based on this, the inspector will not pursue additional enforcement action.

No violations, deviations, unresolved or inspection followup items were identified.

3. Operational Safety Verification (71707, 71710, 42700)

The facility steady state, shutdown, and startup activities were observed as conducted in the plant and from the main control room.

Performance of reactor operators and senior reactor operators, shift engineers, and auxiliary equipment operators was observed and evaluated. Included in the review were procedure use and adherence, records and logs, communications, shift/duty turnover, and the degree of professionalism of control room activities.

Evaluation, corrective action, and response for off normal conditions were examined. This included compliance to any reporting requirements.

Observations of the control room monitors, indicators, and recorders were made to verify the operability of emergency systems, radiation monitoring systems, and nuclear reactor protection systems. Reviews of surveillance, equipment condition, and tagout logs were conducted. Proper return to service of selected components was verified.

a. General

The plant began and ended the report period at essentially full power, although a forced outage occurred from April 28 through May 16, 1993.

b. Unusual Event - Two Diesel Generators Simultaneously Inoperable

On April 27, Diesel Generator (DG) 1-1 was scheduled for preventive maintenance of the air start motors. To facilitate the maintenance activity, DG 1-2 was started and paralleled to the grid. DG 1-2 was loaded to approximately 500 kw when it lost load, was manually tripped, and declared inoperable. DG 1-1 was then verified operable by starting it and paralleling it to the grid.

The licensee administratively declares a DG inoperable when it is paralleled to the grid because under certain degraded grid conditions, the DG voltage regulator may cause the DG to trip and lockout. This would require operator action to reset the lockout condition. When DG 1-1 was paralleled to the grid, it was administratively declared inoperable.

Since both DGs were simultaneous inoperable (DG 1-1 paralleled to the grid and DG 1-2 unable to maintain load), the licensee declared an unusual event and made the notifications required by the emergency plan and 10 CFR 50.72. The licensee subsequently determined that notifications to local officials were not made within 15 minutes because questions from the first official resulted in a communication delay to subsequent officials. The delay was the subject of an internal corrective action document.

The reason that DG 1-2 lost load was because of a defective fuel oil pump. The pump had developed a leak at the shaft seal which permitted air to mix with the fuel oil. The fuel oil pump was repaired and DG 1-2 returned to service.

During trouble shooting activities of DG 1-2, the selector hand switch for the fuel oil filters was found in an abnormal position. With the fuel oil filter removed from service, the licensee verified that the position of the switch did not reduce the flow of fuel oil to the fuel pump. The inspector agreed with the licensee's contention that the position of the selector hand switch did not affect DG operability. The inspector pointed out to the operations manager and again at the exit interview that since the selector switch was not in the preferred position, it was only fortuitous that the as-found position of the fuel oil selector switch did not affect operability.

c. Unusual Event - Plant Removed From Service Due to Excess Water Leakage From Control Rod (CR) Seal Packages

On April 28, the licensee was scheduled to remove the unit from service because leakoff and temperature measurements for two control rod primary coolant system pressure boundary seals were approaching preset administrative limits. The licensee had been monitoring the increased leakoff and temperature measurements since November 1992. Several unsuccessful attempts were made to curb the increasing trend. These included a change in the leakoff sampling frequency and a change in the exercising frequency.

About one-half hour before the commencement of the planned shutdown, the unidentified primary coolant system leak rate was found to exceed the Technical Specification limit of 1.0 gpm by about 0.15 gpm. The licensee performed a second leakrate calculation, declared an unusual event, and started the shutdown at 5:37 p.m. The unit was removed from service at 10 p.m. The licensee secured from the unusual event on April 30 at 2:35 a.m. after the plant entered cold shutdown.

The licensee replaced six control rod drive seal packages, a primary coolant pump seal cartridge, and performed various other maintenance and surveillance activities.

The unit was returned to service on May 16, 1993, at 3:02 p.m. During the post outage heatup, the inspector verified that the control rod drive seal packages and the primary coolant pump seal cartridge parameters were normal.

d. Shutdown Activities

During the outage discussed in the previous paragraph, the inspector made routine tours of the control room. During these tours, the inspector observed that manning requirements were always met, the operators were cognizant of changing plant conditions, the Limiting Condition for Operation (LCO) status board was maintained up-to-date, and the operators were performing assigned tasks in accordance with plant procedures. Several of the activities observed were:

(1) Plant shutdown to hot standby/shutdown and plant cooldown from hot standby/shutdown per General Operating Procedures (GOP) 8 and 9.

(2) Shutdown cooling activities per System Operating Procedure (SOP) 3.

(3) Draining the primary coolant system per SOP 1. This activity was performed in two stages. The first level reduction permitted change-out of the control rod seal packages. The second stage required draining to reduced inventory for change-out of a primary coolant pump seal cartridge.

Prior to commencement of the second inventory reduction, the inspector performed a review of SOP 1 and the licensee's shutdown risk assessment using NRC Temporary Instruction TI 2515/113, "Reliable Decay Heat Removal During Shutdown," as a reference. The inspector verified that both shutdown cooling trains (including mechanical and electrical support systems) were operable, outage activities that could affect shutdown cooling were reviewed and scheduled to minimize the risk to shutdown cooling, and containment integrity was maintained while in reduced inventory.

The inspector attended several shift briefings pertaining to reduced inventory activities. These briefings were conducted by the shift supervisor, were very detailed, and were attended by senior site managers. The shift supervisor clearly discussed the procedural steps required to obtain shutdown cooling and the lessons learned when shutdown cooling was lost at other sites. The site managers stressed the safety significance of shutdown cooling, emphasized the importance of self checking, and accentuated the consequences if shutdown cooling was lost.

The licensee ensured that the majority of the plant staff, not directly involved with reduced inventory activities, were aware of the reduced inventory condition by discussion at the morning meeting. Additionally, reduced inventory was the topic of a news bulletin that received plant wide distribution and was posted throughout the plant.

(4) The inspector routinely reviewed the shutdown risk assessment and verified that outage activities did not create any safety concerns.

(5) Power escalations after synchronization per GOP 5.

(6) Starting and loading of the Diesel Generator per SOP 22.

e. Operator Errors

The inspector reviewed the licensee's corrective and preventive actions for several recent operator errors. These errors were of concern because the number of errors has increased over the past few months. Individually or collectively, none of the errors caused a significant safety problem. The general topic - Personnel Errors by Operators - will be addressed as an unresolved item (Unresolved Item 255/93008-01(DRP)).

As stated in the cover letter, the licensee was requested to respond to this unresolved item by providing an evaluation of the problems and corrective actions. Several of the operator errors that have occurred since the beginning of 1993 are discussed below.

(1) On January 6, 1993, both emergency diesel generators were simultaneously and inadvertently removed from service because an operator incorrectly implemented a switching and tagging order.

(2) On January 12, 1993, an auxiliary operator misaligned air coolers to the iso-phase bus ducts during his shift rounds.

(3) On January 30, 1993, safety injection system surveillance test QO-21 was commenced without two valves in their proper

pre-test alignment. The lineup error was identified, corrected, and the test completed satisfactorily.

(4) On February 24, 1993, feedwater purity building air compressor C-903A was incorrectly removed from service. The error was identified and resolved before the clearance was released to the maintenance crew.

(5) During the March 10, 1993, biweekly control rod drive testing, an operator tested two control rod drives that were not scheduled to be tested. Performance of the test met the testing frequency specified in the Technical Specification but contributed to an unplanned outage.

(6) On March 21, 1993, the spent fuel pool handling machine malfunctioned due to a personnel error. This error was the subject of the enforcement action discussed in the cover letter and discussed in paragraph 8 of this inspection report.

The licensee has implemented several measures to address these problems. They have performed a quality assurance audit of the new self checking and procedure usage policies, issued an operations department news letter outlining the above examples and requesting that operators review their mistakes and think about ways to prevent similar errors in the future, and implemented their progressive disciplinary action policy where appropriate.

No violations, deviations, or inspection followup items were identified. One unresolved item was identified.

4. <u>Maintenance</u> (62703, 42700)

Maintenance activities in the plant were routinely inspected, including both corrective maintenance (repairs) and preventive maintenance. Mechanical, electrical, and instrument and control group maintenance activities were included as appropriate. The focus of the inspection was to assure the maintenance activities reviewed were conducted in accordance with approved procedures, regulatory guides and industry codes or standards, and in conformance with Technical Specifications. The following items were considered during this review: the Limiting Conditions for Operation were met while components or systems were removed from service; approvals were obtained prior to initiating the work; activities were accomplished using approved procedures; and post maintenance testing was performed as applicable.

The following work order (WO) activities were inspected:

a. WO 24204236 Motor driven auxiliary feedwater pump P-8A coupling inspection

b. WO 24300882 Lubrication of pump coupling of motor driven auxiliary feedwater pump P-8A

c. WO 24301093 Troubleshoot motor and controls of west engineered safeguards sampler pump P-1811

d. WO 24301300 Repair of diesel generator 1-2 fuel oil pump

No violations, deviations, unresolved or inspection followup items were identified.

5. Surveillance (61726, 42700)

The inspector reviewed Technical Specifications required surveillance testing as described below, and verified that testing was performed in accordance with adequate procedures. Additionally, test instrumentation was calibrated, Limiting Conditions for Operation were met, removal and restoration of the affected components were properly accomplished, and test results conformed with Technical Specifications and procedure requirements. The results were reviewed by personnel other than the individual directing the test, and deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel.

The following activities were inspected:

a. MI-1 NI Power Range, Rod Drop Alarm, Flux - Delta T Calibration System

b. MI-6 Area Monitor Operational Check

c. MO-7A-1 Emergency Diesel Generator 1-1

d. QO-8 ESS Check Valve Operability Test and High Pressure Safety Injection Flow Indicator Verification

e. QO-6 Cold Shutdown Valve Test Procedure (Including Containment Isolation Valves)

f. QO-10 Containment Spray Pump Operability Test

The licensee evaluated surveillance activities QO-6, 8, and 10 and concluded that performance of selected sections of these tests could have a negative effect on shutdown cooling. As a result of this review the licensee delayed performance of these tests until the end of the forced outage. At that time the steam generators were available as an alternate source of cooling.

No violations, deviations, unresolved, or inspection followup items were identified.

6. Engineering and Technical Support (37700, 92705)

The inspector monitored engineering and technical support activities at the site and, on occasion, as provided to the site from the corporate office. The purpose was to assess the adequacy of these functions in contributing properly to other functions such as operations, maintenance, testing, training, fire protection, and configuration management.

a. Defective Anti-rotational Keys in Motor-Operated Valves (MOV)

A Region III Technical Issue Summary documented the discovery of broken anti-rotational keys in some Velan MOVs due to defective material. The inspector met with the MOV system engineer and his supervisor to determine if this problem had occurred at Palisades. The licensee was aware of the problem and had previously contacted the vendor to confirm that the key material may be defective. Additionally, the licensee was informed that the anti-rotational key will perform adequately during static testing but may fail during dynamic testing.

Palisades has 13 Velan MOVs that have a similar "L" shaped anti-rotational key installed in the mounting flange. The licensee suspects that all 13 valves have the same key material. All of the valves were installed in safety related systems. Since the problem was identified, the licensee has performed the following:

(1) Work Requests were initiated to inspect and replace the keys on the 13 MOVs.

(2) The licensee contacted Velan to discuss key material changes and the reason of the failure.

(3) A functional equivalent substitution to authorize the key replacement was initiated.

(4) An evaluation of plant performance and shutdown risk plant performance was initiated.

(5) The licensee contacted other utilities to obtain additional information.

(6) The system engineer performed a plant-wide search to determine if the problem may be applicable to valves that are not addressed by the licensee's GL 89-10 program.

The inspector found that the licensee's prompt response to this issue was a strength.

b. Fiberglass Piping

The inspector interviewed the balance of plant system engineering section head to determine if fiberglass piping was installed at Palisades. This was in response to a broken fiberglass service water pipe event at another plant. The information was provided to the Nuclear Reactor Regulation License Project Manager.

Fiberglass piping was installed in the cooling tower distribution system that runs along the top of each tower. The piping is valved to evenly distribute water to each cell of each tower. Pipe sizes range from 24 inches to 60 inches.

Sodium hypochlorite storage tanks T-18A and T-18B, as well as circulating water treatment tank T-44, are all fiberglass reinforced storage tanks. The piping from these tanks is non-fiberglass.

c. NUREG/CR-5822, "Analysis of Thermal Mixing and Boron Dilution in Pressurized Water Reactor (PWR)"

The inspector responded to a Region III request for information regarding the potential for inadvertent reactivity insertion from a reactor coolant pump restart. NUREG/CR-5822 identified a scenario which could result in a reactivity insertion. The inspector was asked to determine if the licensee considered this potential transient and has implemented measures to prevent it.

The inspector found that the licensee has procedures in place that should prevent such a transient. Standard operating procedure SOP 2A, "Chemical and Volume Control System Charging and Letdown: Concentrated Boric Acid," required that if an operating primary coolant pump trips, then any dilution in progress be immediately stopped. The procedure also required that at least a primary coolant pump or a shutdown cooling pump be in operation during a change in boron concentration.

The transient was also addressed in the emergency operating procedures. The Reactor Trip Recovery Procedure stated that if less than four primary coolant pumps are operating, then commence emergency boration and establish a 3.75 percent shutdown margin.

No violations, deviations, unresolved, or inspection followup items were identified.

7. Reportable Events (92700, 92720)

The inspector reviewed the following Licensee Event Reports (LERs) by means of direct observation, discussions with licensee personnel, and review of records. The review addressed compliance to reporting requirements and, as applicable, that immediate corrective action and appropriate action to prevent recurrence had been accomplished.

The five reactor trips that were documented in LERs 255/92034, 92035, 92037, 92038, and 92039 are discussed in the three subsequent paragraphs. Four trips were the result of modifications to improve plant performance and the fifth was the result of plant aging. The number of reactor trips contributed to a decline in plant performance as indicated on NRC's Quarterly Performance Indicators. This decline was the subject of two management meetings. The first was held on October 30, 1992, in the NRC's Region III office. The second was held in the NRC's Washington DC office on November 30, 1992.

a. (Closed) LERs 255/92034, 255/92035, and 255/92039: Reactor Trip caused by Loss-Of-Load Resulting From Unstable Voltage to the Turbine Control Computers.

Three full power turbine/reactor trips occurred on July 1, July 24, and October 30, 1992, when in-plant voltage transients momentarily interrupted power to the turbine control computers. The turbine control computers were designed to prevent a turbine over-speed event. Whenever the computers are unable to monitor turbine speed, the turbine is tripped automatically. The reactor trips and subsequent plant responses were discussed in Inspection Reports 255/92018(DRP), 255/92022(DRP), and 255/92023(DRP).

The turbine control computers were added during the previous refueling outage to enhance turbine reliability. The modification provided redundant computers, redundant power supplies, and the capability to switch from a primary to an alternate computer or power supply. Additionally, the licensee elected to keep the turbine control computer power supplies independent of safety related power supplies. This assured that safety and non-safety related power supplies were not cross connected.
However, this modification resulted in unfiltered and unstable power to computers that were designed to function with very stable power supplies. The modification package documented that problems with similar turbine control computers installed at other plants were evaluated and resolved before the modifications were implemented. However, the installation package did not document that differences in the installed configuration were evaluated.

(1) <u>The July 1, 1992, turbine/reactor trip</u>

The licensee did not recognize the significance of a switchyard electrical transient that preceded the turbine/reactor trip. The transient was caused by a ground fault that cleared several cycles before the turbine trip. The licensee concluded that the transient had cleared but incorrectly concluded that the transient did not cause the computer fault. The licensee concluded that connecting cables in the control computers data processing units were not properly connected, vibrated loose, and faulted the computers. This tripped the turbine and generated a loss-of-load signal to the reactor protective system. The licensee was able to duplicate the computer fault by "wiggling" the connections. The control computers were located in a high vibration area, however, as shown in the subsequent paragraphs, the most probable cause was the voltage transient. The licensee acknowledged and documented this conclusion in a subsequent LER.

The licensee did not correctly identify the root cause but may have identified and may have resolved a future turbine trip mechanism - the loose connectors.

(2) The July 24, 1992, turbine/reactor trip

The licensee concluded that this trip was caused by an in-plant voltage transient generated during the performance of an unrelated surveillance test. The transient caused a momentary loss of the turbine control computers. The licensee upgraded the computer power supply switching cards and added a single uninterruptable power supply (UPS) to the power feeds for selected turbine control computers. The UPS provided a filtered power supply that could control rapid voltage transients and provided a limited battery supply to permit operator response if power was lost to the control computers. The licensee's post modification testing confirmed that the UPS resolved rapid transients and helped manage a total loss of power to the control computers.

The licensee did not consider or evaluate the consequences of a degraded or slowly degrading power supply to the control computers. Failure to test for degraded voltage contributed to the October 30, 1992, reactor trip.

(3) The October 30, 1992, turbine/reactor trip

The licensee determined that the supply voltage transformer to the DEH computers was consequently degraded by approximately 10 percent because the taps had been improperly set. This caused the UPS internal logic to remove the UPS from service and permit degraded and unfiltered power to the control computers. The degraded and unfiltered power supply eventually faulted the computers and caused the trip.

This was resolved by changing the transformer tap setting which increased the nominal supply voltage. Additionally the licensee revamped the power scheme to the control computers by replacing the internal computer power supplies, increasing the battery size to the UPS, installing voltage regulating transformers to the control computers not powered by the UPS, and rerouting the power feeds to provide redundant power supplies to the control computers.

(4) <u>The inspector concluded that</u>:

(a) The licensee had evaluated and resolved problems encountered at other facilities prior to installation of the turbine control computers. However, the licensee did not evaluate the differences between the installed Palisades configuration and the installed configuration at other power plants.

(b) The licensee did not evaluate the power scheme prior to installing the turbine control computers or prior to installation of the UPS.

(c) The current power scheme was redundant from load center 14. The licensee recognized that the power supplies to load center 14 were not redundant. The licensee was evaluating modifications to enhance the power scheme.

(d) The operator and plant response to the reactor trips were uncomplicated which speaks highly of the material condition of the plant and the training provided to the operators.

(e) The current power scheme was tested several times when unplanned external line faults caused internal plant transients. In all cases the turbine control computers remained in service.

(f) The licensee reorganized the Design Engineering Department following the 1990 steam generator replacement outage and relocated the department to the site. The inspector was unable to determine if the reorganized department inherited the DEH modification or was the sole sponsor of the modification.

b. (Closed) LER 255/92037: Reactor Trip Caused by Low Steam Generator Water Level Resulting from a Broken Air Line on a Main Feedwater Regulating Valve.

A full power plant trip occurred on August 14, 1992, when the feedwater regulating valve to the "A" steam generator malfunctioned. The malfunction was the result of a broken air line to the valve actuator. This permitted the valve to drift partially shut, resulted in a reduction in feedwater flow, reduced water level in the "A" steam generator, and a reactor trip. The air line was replaced and the air lines to other valves located in the turbine building were inspected before returning the unit to service. The reactor trip and subsequent plant response were discussed in Inspection Report 255/92022(DRP).

c. (Closed) LER 255/92038: Reactor Trip Caused by a Loss of the Preferred Bus Y-20 Coincident with a Blown Fuse in a Second Channel of the Reactor Protective System.

The full power plant trip on August 25, 1992, resulted from a malfunction of preferred ac bus Y-20, coincident with a blown fuse in another portion of the reactor protective system (RPS). The reactor trip and subsequent plant response were discussed in Inspection Report 255/92022(DRP).

The malfunction was caused by an incorrectly wired transformer in Y-20. The wiring error occurred during corrective maintenance performed in 1986 and resulted in accelerated aging of Y-20.

The RPS consists of six logic ladders (AB, AC, AD, BC, BD, and CD) representing the 2 out of 4 logic combinations. Each logic ladder has two auctioneered power supplies which are powered from separate preferred ac power sources. One side of the BC logic matrix ladder is powered by the Y-20 bus while the other side is powered by the Y-30 bus. A fuse blew in the BC matrix power supply which is powered from the Y-30 inverter. The exact time that the fuse blew is unknown. Since the power supplies are auctioneered, either matrix power supply can be in service. When a matrix power supply is out of service it is not annunciated. The fuse was determined to be undersized and could have blown when the power supply was first placed into service following the refueling outage, after some duty time, or when Y-20 malfunctioned.

The power supplies (12 total) were replaced during the previous refueling outage as part of a material condition improvement. The licensee's post trip evaluation determined that the undersized fuse was installed at the vendor's manufacturing facility. The fuses in the other power supplies were inspected and found to be properly sized.

The site inspection staff and a Region III inspector evaluated this LER and concluded the following.

1) The licensee's receipt inspection requirements for the logic ladder power supplies were evaluated. The inspector concluded that the pre- and post- installation activities in combination with the vendor's quality inspection program were adequate to confirm operability of the power supplies.

   The inspector concluded that disassembly of the power supplies during receipt inspection was not required.

2) The current maintenance programs are significantly different from the program that was in place when the Y-20 transformer wiring problems occurred approximately 8 years ago. The difference included the use of system engineers, use of work

order planners, and implementation of a computerized work order system.

d. (Closed) LER 255/92-036: Inadvertent Isolation of Fuel Oil to One Cylinder of the 1-1 Emergency Diesel Generator.

The subject of this LER was reviewed in Inspection Reports 255/92022(DRP) and 255/92023(DRP).

(1) General

On August 2, 1992, the licensee was performing MO-7A-1, "Emergency Diesel Generator (DG) 1-1 (K-6A)," when they observed that cylinder 8R indicated a low exhaust temperature of about 170 degrees F. This was well below the expected temperature of approximately 880 degrees F. The licensee declared the DG inoperable and dispatched an operator to check the fuel rack to cylinder 8R. An auxiliary operator found the fuel rack to cylinder 8R was "latched out" or "shut off." The auxiliary operator opened the latch and the surveillance test was completed satisfactorily.

The fuel racks were locked out 33 days earlier to perform compression testing per MO-7A-1. Following the compression test, the licensee ran the DG and documented that the exhaust temperatures were correct. Additionally, the DG was tested at least twice following two subsequent reactor trips. Although this testing did not require verification of cylinder exhaust temperatures, the licensee's records did not document any operational problems.

(2) Potential Duration of Inoperability

The licensee back dated the inoperability a total of 33 days. This was the elapsed time since the compression testing was satisfactorily performed on June 30, 1992. This meant that the inoperability condition could have existed longer than the allowable outage time of Technical Specification 3.7.2.i. Technical Specification 3.7.2.i permitted an out-of-service time of 7 days, or required a plant shutdown.

(3) Root Cause and Corrective Action

The most likely cause for this event was a failure to implement post compression restoration steps of MO-7A-1. Normally the individual fuel rack latches are stored in the "6 o'clock" position. The latch for this fuel rack may have stuck in an "intermediate or 12 o'clock" position. The licensee confirmed that it is possible for fuel rack latches to stick in an intermediate position. Vibrations may have caused the latch to vibrate from an intermediate position into the "lockout or 3 o'clock" position, thereby isolating the fuel to cylinder 8R. This apparently occurred at some time during the four times that the DG was operated since the compression test.

The licensee's corrective actions included revising MO-7A-1 and MO-7A-2, "Emergency Diesel Generator 1-2 (K-6B)." The revisions included specific directions for proper fuel control rack latch disengagement and caution statements explaining consequences of not completing this properly. Operator training was also held.

(4) Previous Fuel Rack Operability Problems

While evaluating the licensee's past performance to address the criteria of Section VII.B of the enforcement policy (10 CFR 2 Appendix C), the inspector revisited another cylinder exhaust temperature problem which was documented in NRC Inspection Report 255/90039(DRP).

On December 18, 1990, maintenance mechanics were performing preventive maintenance on DG 1-2 when a mechanic unknowingly placed the latch to cylinder 2L in the lockout position. He thought he was placing it in the correct position.

The licensee found this condition during post-maintenance testing and prior to performing the operability run. Operators placed the latch in the correct position and later successfully completed the operability run.

The inspector determined that the corrective actions for the December 18, 1990, event were adequate. They included training of the maintenance mechanics and posting of warning signs in both DG rooms. The inspector found the corrective actions for the 1990 event could not reasonably have prevented the 1992 event. To date, there have been no other instances of mechanics incorrectly positioning fuel rack linkages.

(5) Potential Enforcement Action

This event was evaluated for escalated enforcement by Region III and the Office of Enforcement. Their evaluation determined that escalated enforcement would not be pursued because the post maintenance test performed subsequent to the compression test did verify operability of the fuel racks. Because of that test, the length of time that the DG was inoperable prior to discovery of the fuel rack mispositioning was indeterminate. Therefore, it was not possible to establish that there was any violation of Technical Specification 3.7.2.i.

A violation for not implementing the restoration steps of MO-7A-1 was not cited since the criteria specified in Section VII.B.2 of the "General Statement of Policy and Procedures for NRC Enforcement Actions," (Enforcement Policy, 10 CFR Part 2, Appendix C), were satisfied.

e. (Closed) LER 255/92017: Fuel Assembly Partially Removed From the Core During Upper Guide Structure Lift.

On February 29, 1992, while removing the upper guide structure from the vessel, a fuel assembly at location Z-11 remained attached to the bottom of the upper guide structure. The licensee declared an unusual event, stopped the lift, and implemented contingency plans to secure, free, and lower the fuel assembly back into the core. The licensee subsequently determined that the upper guide structure alignment pins for location Z-11 were bent and caused the fuel assembly to stick during lifting. The pins were straightened before the upper guide structure was installed during vessel restoration activities.

NRC inspections were documented in Inspection Reports 255/92006(DRP) and 255/92015(DRP). In addition, this event was the subject of enforcement action documented in Inspection Report 255/92015(DRP).

f. (Closed) LER 255/91020: Inadequate Documentation - Environmental Protection Plan.

On November 12, 1991, the NRC identified to the licensee that they had not properly documented Unreviewed Environmental Question (UEQ) determinations for plant modifications. This was the subject of enforcement action documented in Inspection Report 255/91024(DRP).

The licensee subsequently determined that UEQ determinations had not been properly documented since February 1987. In February 1987, Administrative Procedure 4.22, "Nonradiological Environmental Program," was revised and deleted the form which provided documentation of an UEQ evaluation in compliance with the Environmental Protection Plan (EPP).

The plant had relied on the Corporate Environmental Department to determine if a test, change, procedure or modification could involve an UEQ. The licensee stated that the Corporate Environmental Department was responsible for compliance with federal, state, and local environmental regulations. However, they were not trained or knowledgeable in nuclear regulatory compliance.

The cause of this event was a procedure inadequacy. The licensee's corrective actions were: (1) suspend processing of environmental impact reviews until Administrative Procedure 4.22

was revised to include steps to ensure the environmental requirements of the EPP are met for determining the involvement of a UEQ; and (2) review all evaluations of the involvement of UEQs performed since February 1987 and perform required evaluations as necessary.

The inspector verified that Administrative Procedure 4.22 was properly revised. He also reviewed several design changes and confirmed that required evaluations were performed.

g. (Closed) LER 255/92004 (original and supplement dated March 6, 1992, and May 15, 1992): Potential Loss of Containment Integrity Due to the Failure of the Emergency Escape Airlock Equalizing Valve.

The LER was superseded by LER 255/93003.

The licensee concluded that the equalizing valve stuck partially open because the valve stem lubricant partially dried and became tacky. The licensee cleaned the equalizing valve during the 1992 refueling outage and lubricated the stem with the vendor recommended lubricant. The equalizing valve was returned to service and operated properly during post maintenance testing. The equalizing valve failed a second time during airlock testing performed on March 6, 1993. The second failure was discussed on LER 255/93003. That LER remains open to be evaluated when the valve is disassembled during the upcoming 1993 refueling outage.

One non-cited violation was identified. No deviations, unresolved, or inspection followup items were identified.

8. <u>Spent Fuel Pool Handling Machine (SFHM) Malfunction Due to a Personnel Error</u>

a. <u>General</u>

During preplanned moves of spent fuel (to support the up-coming 1993 refueling outage), the SFHM malfunctioned due to a personnel error. The malfunction occurred after an operator had inappropriately used the SFHM "Override Key Switch" in an unsuccessful attempt to seat and ungrapple a fuel assembly in a storage location. When the operator attempted to remove the fuel assembly from the storage location, several inches of the main cable unwrapped from the SFHM drum and wrapped around the motor shaft. The cable remained attached to the drum but allowed the fuel assembly to drop approximately six inches. The fuel assembly was suspended approximately six inches above the bottom of the spent fuel pool (SFP). This occurred at approximately 2:30 a.m. on March 21, 1993. The licensee verified that the fuel assembly was not damaged and confirmed that there was no release of radioactivity.

The shift supervisor directed that power to the SFHM be secured and he suspended activities in the SFP. The shift engineer verified that the emergency plan had not been entered and that a 10 CFR 50.72 telephone notification was not required. A system engineer, a system engineering section chief, the shift supervisor, and a field representative for the refueling machine examined the configuration and concluded that the fuel assembly was in a safe configuration.

b. <u>NRC activities</u>

The senior resident inspector was notified at approximately 4:30 a.m. and responded to the site. He attended the licensee's technical briefings, interviewed the operator who performed the fuel moves, reviewed the procedure controlling the activity, and independently verified that the fuel assembly was in a safe configuration. Based on interviews with the operator performing the activity, the inspector concluded that personnel error was a primary cause. The inspector informed the operations superintendent at approximately 9:30 a.m. that personnel error was a primary cause of this event.

c. <u>Spent Fuel Pool Handling Machine Operating Information</u>

The SFHM can be operated in semi-automatic mode, in manual mode with the computer interlocks functioning, or in manual mode with the computer interlocks and monitoring capabilities bypassed.

The manual mode (with the computer in service) permitted manual operation of the bridge, trolley and hoist within operating zones that were preprogrammed into the computer. The hoist is only operated in manual and is normally operated with the computer controlling the speed of the hoist, monitoring the load, the position of the fuel assembly, the upper grapple operating zone (UGOZ) and lower grapple operating zone (LGOZ).

The semi-automatic mode permitted computer control of the direction, speed, and operating zones of the bridge and trolley.

The system (bridge, trolley, and hoist) can be operated without the computer by use of the "Override Key Switch." The "Override Key Switch" bypassed all of the limit switches and operating zones. This mode of operation provided the operator with a means to safely store a fuel assembly in the event of a computer malfunction. The operating procedures for the SFHM were contained in Section 7.6 and Attachment 5 of System Operating Procedure (SOP) 28, "Fuel Handling System."

d. <u>Event and Violation</u>

While lowering fuel assembly XF-4 into storage location QW-35, the operator observed normal weight of 1300 to 1400 pounds. When the fuel assembly was approximately one and one half inches from seating, the underload interlock activated and automatically stopped the hoist. In this case, the underload condition identified that the fuel assembly alignment pins were resting on the bottom of the storage location but not entering the alignment holes of location QW-35. Two alignment pins (approximately one and one half inches in length) are located on the bottom of each fuel assembly. Each storage location has four alignment holes.

A fuel assembly was considered properly seated if it traveled the programmed distance and the alignment pins entered the alignment holes. When these conditions were established, the lower grapple operating zone (LGOZ) position switch would activate, the underload condition would activate the cable slack limit switch, and permit a manual ungrappling of the fuel assembly.

When the underload conditicn was received, the operator raised the fuel assembly several inches, lowered the fuel assembly, and attempted to reseat the fuel assembly. The underload condition was received a second time. At that time, the operator activated the "Override Key Switch" and unsuccessfully attempted to lower the fuel assembly to the LGOZ and seat the bundle. The operator then removed the fuel assembly from storage location QW-35, realigned the fuel assembly, and attempted to reseat the fuel assembly. The operator was unable to seat the fuel assembly and again attempted to seat the fuel assembly by use of the "Override Key Switch." The operator used the "Override Key Switch" without permission of the shift supervisor which was a violation of step 1 to Attachment 5, "Key Operated Override Switch Operating Guidelines" of SOP 28. Step 1 stated, "Use of key operated override switch shall be at the direction of an SRO." Technical Specification 6.8.1.b required implementation of procedures for fuel handling activities. SOP 28 implemented this requirement. Failure to comply with step 1 of Attachment 5 to SOP 28 is a violation of Technical Specification 6.8.1.b (Violation 255/93008-02(DRP)).

The operator removed the fuel assembly from the storage location, rotated the fuel assembly 90 degrees, and successfully seated the bundle in storage location QW-35. The operator was not able to ungrapple the fuel assembly. The operator then attempted to remove the fuel assembly with the intention of rotating the fuel assembly 90 degrees. When the operator raised the fuel assembly approximately 12 inches, several inches of the main cable unwrapped from the drum and wrapped several times around the drive shaft. This permitted the fuel assembly to drop approximately six inches and become suspended approximately six inches above the bottom of the SFP.

e. Review of SOP 28

The inspector reviewed SOP 28 to determine if the procedure was adequate for the activity performed. The inspector found that the procedure was easy to follow and addressed the normal or routine activities. The inspector questioned if the procedure adequately addressed off normal conditions. For example, the procedure was silent with respect to the underload condition.

The procedure required shift supervisor approval prior to use of the "Override Key Switch." It also provided specific criteria pertaining to its use. The violation was based on the operator's failure to obtain shift supervisor permission prior to use of the Override Key Switch. The inspector also noted that if the operator had reviewed the procedure he would have not proceeded because use of the override key switch for this purpose was not addressed.

f. Recovery Activities

The system engineer and the SFHM vendor implemented a course of action that included installation of a clamping device on the main cable, attaching the clamping device (via a chain fall) to the overhead crane, and suspending the cable and fuel assembly from the overhead crane. The cable and drum were manually detensioned and the cable rewound on the drum. The drum was manually tensioned, the clamping device removed, the bundle lowered and the grapple disconnected from the fuel assembly with the aid of a long reach pole.

One violation and no deviations, unresolved, or inspection followup items were identified.

9. Dry Cask Storage Operations (42700, 86700, 37702, 37703, 71707)

a. Decontamination of Multi-assembly Sealed Basket (MSB)

The inspector observed part of the decontamination of the MSB and the MSB Transfer Cask (MTC) according to procedure T-FC-864-01, "Preoperational Test Procedure for Loading and Placing the Ventilated Storage Cask into the Storage." The inspector found that the evolution was properly controlled.

Information provided during the pre-job briefing included precautions, a review of the procedural steps, and identification and responsibilities of key management, technical, and operations personnel. The briefing was effective and met the objectives of the procedure.

The licensee performed this activity to verify that the gap between the MTC and the MSB could be decontaminated to a level less than 2000 cpm/100cm^2 smearable. The decontamination was accomplished by flushing the gap with pure water as it was lifted out of the spent fuel pool. Prior to the flush, the licensee measured the contamination level and found the highest contamination level to be 800 cpm/100cm^2, well below the acceptance criteria. The gap was then flushed to further reduce the contamination levels.

The decontamination of the MSB/MTC gap was the second time this phase of the preoperational procedure was performed. The licensee had a more difficult time decontaminating the MTC/MSB a few weeks earlier. They suspected this was partly due to the extended amount of time (approximately 4 weeks) the MTC/MSB remained in the spent fuel pool during refurbishment of its transport system.

They also suspected a problem with the shim material used to maintain the gap. The shims provide radial alignment of the MSB within the MTC. The licensee changed the shim material from carbon steel to stainless steel. The carbon steel material made decontamination more difficult because corrosion products formed due to reaction with the borated water in the spent fuel pool.

For this current test, the MTC/MSB was in the spent fuel pool about 24 hours prior to its decontamination. During actual fuel loading, the MTC/MSB is expected to be in the spent fuel pool less than 24 hours.

b. <u>Review of Load Distribution System (LDS)</u>

The LDS was installed in the auxiliary building track alley to evenly distribute the load of the ventilated concrete cask to the floor and rooms located below the track alley. Portions of the following calculations were reviewed for compliance with NRC requirements and conformance to licensee commitments:

(1) EA-FC-864-019, "Structural Analysis of the Palisades Track Alley Bridge for VSC-24 Loading Operation," Revision 0, April 22, 1993.

(2) EA-FC-864-020, "Weight Calculation for VCC, MTC, and MSB," Revision 1, February 3, 1993.

(3) EA-FC-864-011, "Evaluation of MSB for Drop Loads for a Hypothetical Drop on the LDS in Track Alley," Revision 2, April 9, 1993.

The analyses were well organized with all assumptions adequately identified. The evaluation of the LDS used a simplified analytical approach and applied bounding loads to the structure at worst-case locations. Point loads were used instead of distributed loads, which added to the conservatism of the calculation. No technical deficiencies were noted during the review of the calculation.

All calculations provided by the transfer cask vendor were reviewed by either cognizant licensee personnel or by a third party consultant. The design verification documentation associated with these reviews indicated that they were technically thorough and comments were adequately resolved. In some cases, alternate calculations were provided to demonstrate the adequacy of the initial calculation and to preclude any questions in that regard.

The NRC inspector briefly examined the installed LDS. Grout installed between the baseplates and road surface had several cracked areas. These were restricted to the portion of the grout that extended beyond the edge of the baseplates and did not appear to be structurally significant. In general the overall workmanship and quality of welding were good. No significant deviations were noted between the design drawings and as-built structure.

During previous inspections that evaluated management of design changes, specifications, and calculations, a number of problems were identified pertaining to control of contractors. The inspector's limited review of the LDS did not identify any additional problems in that area.

c. <u>Public Demonstrations</u>

The inspector observed two public demonstrations protesting against the licensee and one supporting the licensee in the use of dry cask storage. These demonstrations were sponsored by several local groups and held at the plant access road. The demonstrations were peaceful, did not obstruct traffic, received extensive media coverage, and averaged approximately 100 persons per demonstration.

d. <u>Plant Review Committee (PRC)</u>

The inspector attended the May 1, 1993, special PRC meeting. The purpose of the meeting was to summarize the status of the Dry Fuel Storage (DFS) project. This was accomplished by performing a review of the pre-operational test results, summarizing the status of the engineering design packages, discussing the Certification of Compliance (CofC), reviewing the cask loading schedule, and evaluating outstanding licensing issues. The PRC concluded that there were no unreviewed safety questions.

The items discussed were presented to plant management by knowledgeable individuals. These individuals were able to discuss the criteria used to determine whether items were successfully completed. In addition, these individuals were able to discuss the lessons learned and the technical merit of the solutions to any problems.

The inspector verified that the PRC composition met Technical Specification manning requirements and that a voting quorum was present.

e. Legal Activities

On May 4, 1993, a motion for a Temporary Restraining Order was filed by the Attorney General for the State of Michigan, the Lake Michigan Federation, and several land owners with property on the shore of Lake Michigan and adjacent to the plant. The motion was for an order temporarily restraining the NRC from permitting outside storage of spent fuel in VSC-24 casks at the Palisades Nuclear Power Plant. On May 8, 1993, the Judge ruled that his court did not have jurisdiction.

On May 13, 1993, a motion for an Emergency Stay was filed in Appellate Court by the same personnel listed in the preceding paragraph. The Emergency Stay was for an immediate stop work order for using the VSC-24 casks at Palisades Nuclear Power Plant. On May 17, 1993, a panel of three judges at the Federal Court of appeals in Cincinnati, denied the plaintiffs' request for an Emergency Stay.

No violations, deviations, unresolved, or inspection followup items were identified.

10. Loading of the First and Second Dry Storage Casks (71707 and 42700)

Loading of the first cask started shortly after midnight on May 7, 1993. Loading of the second cask started on May 12, 1993. The site NRC inspection staff, NRC inspectors stationed at other plants, Region III-based inspectors, and NRC Headquarters-based inspectors implemented round-the-clock coverage while the licensee loaded fuel in the casks and transported the casks to the storage pad. The activities listed below were inspected.

a. The inspector performed a limited review of the following procedure using 10 CFR 72; the NRC Safety Analysis Report for the Ventilated Storage Cask System - dated April 28, 1993; CofC Number 1007 - dated May 3, 1993; and the American Society of Mechanical Engineers (ASME) Code as references.

(1) FHS-M-32 "Loading and Placing the VSC into Storage"

The inspector verified that the procedure required at least two vacuum pressure drying times with an acceptance criteria of less than 3 mm Hg vacuum, a helium leak test at 22.1 psia, and a final helium pressurization to 14.5 psia. The inspector also verified that the procedure imposed a minimum temperature for lifting the MTC and moving the VCC, and that it imposed a handling height limitation for the MTC and VCC.

During interviews with the site licensing personrel, the inspector was informed that the MSB drain time was administratively reduced based on the ambient temperature of the water in the spent fuel pool.

(2) FHS-M-34 "Unloading the Multi-Assembly Sealed Basket"

(3) FHSO-17 "Multi-Assembly Sealed Basket Loading Procedure"

The inspector interviewed several reactor engineers and the reactor engineering section chief to confirm that the spent fuel assemblies stored in the first two casks met the specification of the CofC. Several questions were asked pertaining to the heat loading of the first cask and integrity of the fuel cladding for the fuel assemblies placed in both casks. The selected heat loading was approximately 12 kW, which was the maximum heat loading available for the required post irradiation time.

The integrity of the fuel cladding was confirmed by visual inspection. Additionally, the fuel assemblies of the first two casks were "sipped" when they were removed from the core in 1986 to ensure they were not leaking.

(4) CC-[illegible]7 "Spent Fuel Pool Chemistry Operating Procedure"

The inspector verified that the procedure required independent determinations of the spent fuel pool boron concentration.

(5) FC-LID and SM-LID "CPCo Welding Procedure"

The inspector reviewed these Weld Procedure Qualification Records (PQRs): FC-LID-A, FC-LID-B, SM-LID-C, and SM-LID-D; these certified material test reports: weld wire E71T-1, heat #32039; weld electrode E-7018-3/32", heat #91232; and weld electrode E-7018-1/8", heat #T20658. In addition the inspector reviewed these welder certifications: Flux core automatic welding (FCAW): 3B, 54; Shielded metal arc welding (SMAW) 3B, 54, 7H, and 7B.

The documents reviewed complied with the requirements of the ASME Code, Section III, NC 4000 and Section IX, 1986 Edition, Summer 1988 Addenda.

b. The inspector attended the prejob and ALARA briefings for the following:

(1) Movement of the MTC/MSB into the spent fuel pool.

(2) Movement of the MTC/MSB from the spent fuel pool to the cask washdown pit.

(3) Welding of the shield lid and structural lid to the MSB.

(4) Movement of the MTC/MSB from the cask washdown pit to the VCC and placement of the MSB into the VCC.

(5) Movement of the VCC to storage.

The pre-job and ALARA briefings were very detailed, conducted by the cognizant supervisor, and attended by senior site managers. The cognizant supervisor clearly discussed the precautions, the procedural steps expected to be accomplished, and identified the responsibilities of the key personnel. The site managers stressed the safety significance of performing the activities in a well controlled manner and stressed the importance of both personnel and plant safety. The inspector noted that the managers did not stress the schedule for completing the activities, and in several instances stated that completion of a quality and safe activity was more important then meeting schedular predictions.

c. The inspector observed the following activities:

(1) Placement of fuel into both MSBs per FHSO 17.

(a) The inspector verified that operators performed dual verification of the identity of each fuel assembly before insertion in the MSB.

(b) The inspector reviewed the official copy of FHSO-17, and noted that the section pertaining to checkout of the refueling machine had been marked "N/A". No explanation for the "N/A" was provided. After some review it was determined that the "N/A" was appropriate because the activity was performed by another procedure. The inspector noted that a <u>brief</u> explanation for the "N/A" appeared appropriate. A brief explanation was subsequently included.

(c) While loading the second cask, the inspector noted that the control room operator directing the fuel moves and maintaining the status board was involved in other duties and appeared to be having difficulty completing all assigned duties. This was discussed with the shift engineer who provided additional assistance.

(2) Welding per FC-LID and SM-LID.

The inspector observed welding of the MSB shield lid for the first MSB. The inspector verified that all welding essential variables were performed in accordance with the applicable welding procedure and that nondestructive examination (liquid penetrant) was performed at the required intervals. The welders were qualified for the process employed, welding materials were identified, and stored in accordance with ASME Code requirements. The licensee assigned a welding engineer to observe and assure that welding activities were performed to applicable requirements.

The inspector concluded that the licensee implemented a welding program that was conservative and assured that the MSB welding was accomplished to the required standards and codes.

The licensee's management attention was duly noted during the inspection.

(3) Movement of the MSB/MTC from the wash down pit to the spent fuel pool, from the spent fuel pool to the wash down pit and to the VCC, and placement of the MSB into the VCC per FHS-M-32. These activities were observed for each MSB.

(4) Movement of both VCCs to the storage pad and placement on the storage pad per FHS-M-32.

d. The inspector concluded the following:

(1) That the procedures reviewed implemented the applicable portions of the certificate of compliance.

(2) Management involvement was highly visible.

(3) The ease that the licensee placed fuel, welded, and transported the casks to storage demonstrated that the preoperational test program was effective and well managed.

No violations, deviations, unresolved, or inspection followup items were identified.

11. <u>Quarterly Management Meeting</u>

A Quarterly Management meeting was held at the Palisades site on May 17, 1993, with the personnel indicated in paragraph 10 in attendance.

The following items were discussed:

- A discussion of the current plant status, including dry fuel storage operations.

- A review of plant operations from November 8, 1992, through April 28, 1993.

- A discussion of the problems, causes, and corrective actions associated with the turbine generator digital electro-hydraulic control system.

- An overview of the 1993 refueling outage scope.

- A discussion of NRC inspections as they impact billing of the licensee and on the licensee's refueling outage.

No violations, deviations, unresolved, or inspection followup items were identified.

12. Persons Contacted

Consumers Power Company

#D. P. Hoffman, Vice President, Nuclear Operations
#G. B. Slade, Plant General Manager
*#T. J. Palmisano, Plant Operations Manager
D. J. VandeWalle, Mech/Civil/Structural Engr. Manager
*#R. D. Orosz, Nuclear Engineering & Construction Manager
*#D. W. Rogers, Safety & Licensing Director
*#K. M. Haas, Radiological Services Manager
J. L. Hanson, Operations Superintendent
R. B. Kasper, Maintenance Manager
* K. E. Osborne, System Engineering Manager
W. L. Roberts, Senior Licensing Engineer
K. A. Toner, Electrical/I&C/Computer Engineering Manager

Nuclear Regulatory Commission (NRC)

#W. L. Forney, Deputy Director, Division of Reactor Projects, RIII
#W. M. Dean, Acting Director, Project Directorate III-1, NRR
#W. E. Scott, Performance and Quality Evaluation Branch, NRR
#W. D. Shafer, Chief, Reactor Projects Branch 2, RIII
#A. H. Hsia, Licensing Project Manager, NRR
#E. R. Schweibinz, Senior Project Engineer, Region III
*#J. K. Heller, Senior Resident Inspector
#D. Passehl, Resident Inspector

*Denotes some of those present at the management interview on May 24, 1993.

#Denotes some of those present at the quarterly management meeting held on May 17, 1993.

Other members of the plant staff, and several members of the contract security force, were also contacted during the inspection period.