Attachment

EMPLOYEE CONCERNS PROGRAMS

PLANT NAME: Palisades LICENSEE: Consumers Power DOCKET #: 50-255

NOTE: Please circle yes or no if applicable and add comments in the space provided.

A. PROGRAM:

1. Does the licensee have an employee concerns program? (Yes or No/Comments)
The licensee has no formal program. They have an unwritten policy that if a concern can not be resolved, through line management, it should be brought to the attention of the Human Resources Dept.

2. Has NRC inspected the program? Report #______

B. SCOPE: (Circle all that apply)

All other Questions N/A since the licensee has no program.

1. Is it for:

   a. Technical? (Yes, No/Comments)

   b. Administrative? (Yes, No/Comments)

   c. Personnel issues? (Yes, No/Comments)

2. Does it cover safety as well as non-safety issues? (Yes or No/Comments)

3. Is it designed for:

   a. Nuclear safety? (Yes, No/Comments)

   b. Personal safety? (Yes, No/Comments)

   c. Personnel issues - including union grievances? (Yes or No/Comments)

4. Does the program apply to all licensee employees? (Yes or No/Comments)

5. Contractors? (Yes or No/Comments)

200088

6. Does the licensee require its contractors and their subs to have a similar program? (Yes or No/Comments)

DF01 0/1

7. Does the licensee conduct an exit interview upon terminating employees asking if they have any safety concerns?
   (Yes <u>or</u> No/Comments)

C. **INDEPENDENCE:**

1. What is the title of the person in charge?

2. Who do they report to?

3. Are they independent of line management?

4. Does the ECP use third party consultants?

5. How is a concern about a manager or vice president followed up?

D. **RESOURCES:**

1. What is the size of staff devoted to this program?

2. What are ECP staff qualifications (technical training, interviewing training, investigator training, other)?

E. **REFERRALS:**

1. Who has followup on concerns (ECP staff, line management, other)?

F. **CONFIDENTIALITY:**

1. Are the reports confidential?
   (Yes <u>or</u> No/Comments)

2. Who is the identity of the alleger made known to (senior management, ECP staff, line management, other)?
   (Circle, if other explain)

3. Can employees be:

   a. Anonymous? **(Yes, No/Comments)**

   b. Report by phone? **(Yes, No/Comments)**

G. FEEDBACK:

1. Is feedback given to the alleger upon completion of the followup? **(Yes <u>or</u> No - If so, how?)**

2. Does program reward good ideas?

3. Who, or at what level, makes the final decision of resolution?

4. Are the resolutions of anonymous concerns disseminated?

5. Are resolutions of valid concerns publicized (newsletter, bulletin board, all hands meeting, other)?

H. EFFECTIVENESS:

1. How does the licensee measure the effectiveness of the program?

2. Are concerns:

   a. Trended? **(Yes <u>or</u> No/Comments)**

   b. Used? **(Yes or No/Comments)**

3. In the last three years how many concerns were raised? ______ Closed? ______ What percentage were substantiated? _______

4. How are followup techniques used to measure effectiveness (random survey, interviews, other)?

5. How frequently are internal audits of the ECP conducted and by whom?

I. ADMINISTRATION/TRAINING:

1. Is ECP prescribed by a procedure? **(Yes <u>or</u> No/Comments)**

2. How are employees, as well as contractors, made aware of this program (training, newsletter, bulletin board, other)?

<u>ADDITIONAL COMMENTS:</u> **(Including characteristics which make the program especially effective or ineffective.)**

The person completing this form please provide the following information to the Regional Office Allegations Coordinator and fax it to Richard Rosano at 301-504-3431.

NAME: TITLE: PHONE #:

____________/______________/_____________ DATE COMPLETED:________

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137-5927

OCT 14 1993

Docket No. 50-255

Consumers Power Company
ATTN: Gerald B. Slade
General Manager
Palisades Nuclear Generating Plant
27780 Blue Star Memorial Highway
Covert, MI 49043-9530

Dear Mr. Slade:

SUBJECT: ROUTINE RESIDENT INSPECTION AT PALISADES NUCLEAR PLANT

This refers to the inspection conducted by Messrs. M. E. Parker, D. G. Passehl, and C. N. Orsini of this office, and B. C. McCabe of the Office of Nuclear Reactor Regulation, from August 13 through September 29, 1993. The inspection included a review of authorized activities for your Palisades Nuclear Generating Facility. At the conclusion of the inspection, the findings were discussed with those members of your staff identified in the enclosed report.

Areas examined during the inspection are identified in the report. Within these areas, the inspection consisted of a selective examination of procedures and representative records, interviews with personnel, and observation of activities in progress.

During this inspection a cooldown of the primary coolant system occurred that exceeded the limit stated in the technical specifications. This is of concern because it appears to have been caused by operator inattentiveness, compounded by an inadequate or ineffective primary system cooldown procedure. It also suggests that you continue to experience operator errors that were identified as a previous weakness. An unresolved item was identified for this issue.

No violations of NRC requirements were identified during the course of this inspection.

In accordance with 10 CFR 2.790 of the Commission's regulations, a copy of this letter and the enclosed inspection report will be placed in the NRC Public Document Room.

9310250096

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

Original signed by

T. Kobetz, Acting Chief
Reactor Projects Section 2A

Enclosures:
1. Inspection Report
   No. 50-255/93021(DRP)
2. Attachment 1
   TI2500/028 Questionnaire

cc w/enclosure:
David P. Hoffman, Vice President
  Nuclear Operations
David W. Rogers, Safety
  and Licensing Director
OC/LFDCB
Resident Inspector, RIII
James R. Padgett, Michigan Public
  Service Commission
Michigan Department of
  Public Health
Palisades, LPM, NRR
SRI, Big Rock Point

bcc: Public

yes

RIII	RIII	RIII
Orsini 10/13/93	Pirok	Kobetz 10/14/93

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

T. Kobetz, Acting Chief
Reactor Projects Section 2A

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Palisades, LPM, NRR
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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-255/93021(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, Michigan

Inspection Conducted: August 13 through September 29, 1993

Inspectors: M. E. Parker B. C. McCabe
D. G. Passehl C. N. Orsini

Approved By: T. Kobetz, Acting Chief
Reactor Projects Section 2A

10/14/93
Date

Inspection Summary

Inspection from August 13 through September 29, 1993
(Report No. 50-255/93021(DRP))

Areas Inspected: Routine, unannounced inspection by resident and regional inspectors of actions on previously identified items, licensee event report followup, followup of events, operational safety verification, maintenance, surveillance, temporary instruction 2500/028, and a management meeting. No Safety Issues Management System (SIMS) items were reviewed.

Results: No violations or deviations were identified in any of the nine areas inspected. One unresolved item was noted and is described in paragraph 5.

The strengths, weaknesses, and Inspection Followup Items are discussed in paragraph 1, "Management Interview."

~~9310250095~~

DETAILS

1. Management Interview (71707)

The inspectors met with licensee representatives (denoted in paragraph 11) on September 29, 1993, and informally throughout the inspection period to discuss the scope and findings of the inspection activities. The inspectors also discussed the likely informational content of the inspection report, including the attachment, with regard to documents or processes reviewed by the inspectors. The licensee did not identify any such documents or processes as proprietary.

Highlights of the exit interview are discussed below:

a. Strengths noted:

(1) Plant housekeeping improved during the period.

b. Weaknesses noted:

(1) Operator inattentiveness during plant cooldown that led to an excessive cooldown rate

(2) Design Control. Several issues that involve inadequate design control were:

- Containment high pressure and high radiation relays were found to be inoperable

- Pressurizer level indicator powered from a nonsafety-related power supply

- Inoperable engineered safeguards room cooler fans

(3) Offsite Dose Calculation Manual effluent concentration activity limit for Safety Injection and Refueling Water Tank was exceeded.

2. Actions on Previously Identified Items (92701, 92702)

a. (Closed) Inspection Followup Item 255/92018-02(DRP): Evaluate Temporary Shielding Installation On Safety Related Piping

During a review of the licensee's program for installing temporary lead shielding, the inspector questioned the adequacy of installation Number 63, which consisted of six lead blankets wrapped around a section of the safety injection and refueling water (SIRW) tank outlet piping. The concern was whether the shielding should have been considered a temporary modification and whether a dynamic load analysis should have been performed.

The licensee's temporary modification procedure specifically excludes lead shielding, which is covered by a separate procedure, namely Administrative Procedure 7.14, "Temporary Shielding Program." This installation met the requirements of that procedure. The lead blankets were installed during an outage when the system was inoperable, and a static load analysis was performed and was acceptable. A dynamic load analysis was performed later when the system was declared operable and likewise was acceptable.

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

3. Licensee Event Report Followup (92700, 92720)

The inspectors reviewed the following Licensee Event Report (LER) 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.

a. (Closed) LER 255/92003: Inadvertent Actuation of the Emergency Diesel Generators due to Undervoltage Condition: During a plant startup in December 1991 with the reactor at 35 percent power, the second heater drain pump was started. At 800 horsepower, the heater drain pumps are the largest loads on the 2400 VAC system. The starting transient caused the voltage on the 2400 VAC system to drop below the setpoints of the "second level" undervoltage relays on the safeguards buses 1C and 1D.

These "second level" undervoltage relays have a built-in 0.5 second time delay, after which both emergency diesel generators (EDGs) receive a start signal. If a bus undervoltage exists after an additional six seconds, then the respective incoming bus circuit breaker will be tripped and a bus load shed will be initiated. In this instance, both EDGs started due to the bus undervoltage, but since the voltage recovered within six seconds, no further automatic actions were initiated.

The cause of this event was failure to properly calculate cable impedance during design and testing of the plant's new safeguards transformer and associated 2500 feet of underground cable. The cable was installed as part of the modification which also installed the new safeguards transformer.

The safeguards transformer incorporates an automatic under-load tap changer to maintain 2400 VAC on the 1C and 1D buses. The tap changer uses a line drop compensator setting to account for the cable impedance and voltage drop between the transformer and the bus supplied by the transformer. Because of the incorrect calculated cable impedance, the compensator setting was also incorrect and resulted in increased voltage drops during motor starting conditions.

The licensee completed a number of corrective actions that included proper evaluation and setting of the compensator, verification of proper cable impedance, verification of bus undervoltage relay actuation setpoints, periodic tap changer compensator setting checks, and evaluation of post modification testing requirements. The corrective and preventive actions appeared satisfactory since no similar problems have been observed to date.

b. (Closed) LER 255/92004: Loss Of Containment Integrity Due To The Failure Of The Emergency Escape Airlock Equalizing Valve: On January 7, 1992, the plant was at full power when the licensee found that containment integrity was breached twice for a short period of time while performing a leak rate test on the containment escape airlock. This occurred on January 6, 1992, when the auxiliary operator opened the escape air lock outer door to exit the escape air lock following the inspection of the inner door, and again when maintenance personnel opened the outer door to enter the escape air lock and install the strongbacks on the inner door

The root cause of this event was an inner door equalizing valve which was stuck in the open position. It was determined that lubricant applied to the valve stem dried out and became tacky, causing the valve to stick during operation.

To prevent recurrence, the maintenance procedure for the airlocks was revised to check the condition of the equalizing valve following routine surveillances. This valve was subsequently repaired during the 1992 refueling outage.

The licensee performed an analysis to quantify the amount of activity released to the environment. The analysis showed a total of 1.35E-2 curies were released. This constituted 3.27E-3 percent of the total maximum permissible concentration as defined in the Offsite Dose Calculation Manual. Therefore, a significant radiological source term was not present at the time of this event.

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

4. Followup of Events (93702)

During the inspection period, the licensee experienced several events, some of which required prompt notification of the NRC pursuant to 10 CFR 50.72. The inspectors pursued the events onsite with licensee and/or other NRC officials. In each case, the inspectors verified that the notification (if required) was correct and timely, that activities were conducted within regulatory requirements, and that corrective actions would prevent recurrence. The specific events are as follows:

August 4, 1993	Containment high pressure/containment high radiation relays found to be inoperable.
August 4, 1993	Pressurizer level indicator LIA-0102A found powered from nonsafety-related bus.
August 9, 1993	Inadequate Emergency Diesel Generator Fuel Oil Supply.
August 19, 1993	Degraded neutron absorbing plates in spent fuel pool.
August 24, 1993	Inoperable engineered safeguards room cooler fans.
August 30, 1993	Offsite Dose Calculation Manual effluent concentration activity limit for Safety Injection and Refueling Water Tank exceeded.
September 16, 1993	Unisolable through wall leak on pressurizer to power operated relief valve outlet nozzle.
September 17, 1993	Excessive cooldown rate during depressurization. (see paragraph 5 for further details)

The following are brief summaries of the events. The inspector will evaluate corrective actions for the events when the respective LERs are reviewed.

a. On August 4, 1993, the containment high pressure actuation relays failed during performance of a surveillance test. Subsequent analysis found that the cause of the failure was an inadequately designed relay, in that the closing coil was not sized per vendor specifications. The relays are manufactured by Clark, Model No. 5U12-76.

   The licensee found that the design discrepancy applied to other relays of this make and model having a certain contact configuration. Specifically, all the Clark Model No. 5U12-76 relays having nine or more normally closed contacts had undersized coils.

   The only other system where these relays are used is in the containment high radiation isolation actuation circuitry. As a result, the containment high radiation isolation system was declared inoperable and refueling activities were immediately halted. The licensee closed all automatic containment isolation valves as a precaution.

All the undersized relays in both the containment high pressure and high radiation systems were replaced with relays having properly sized coils.

b. On August 4, 1993, the licensee determined that pressurizer level indicator LIA-0102A was powered from a non-safety related AC power supply. This condition was discovered during the review of electrical prints in support of the ongoing Configuration Control Program (CCP) drawing review. The licensee plans to power this indicator from the correct AC power source prior to the end of the 1993 refueling outage.

c. During a review of emergency diesel generator (DG) fuel oil consumption rates, the licensee discovered that the rate of DG fuel oil consumption was greater than previously calculated by a small percentage. Additionally, there were inconsistencies between fuel oil consumption rates stated in the technical specifications, final safety analysis report, and systematic evaluation program topic VII-3.

The licensee has placed long and short term actions in place to resolve this issue. The minimum fuel oil level set-point has been raised to ensure an adequate fuel oil supply. The licensee also plans to correct the inconsistencies between the various documents. A satisfactory time line is in place to complete these actions.

d. On August 16, 1993, the licensee identified potential degradation of the neutron absorber material in the spent fuel pool racks. The Palisades plant has high-density fuel storage racks installed in the spent fuel pool which incorporate boron impregnated polymer sheets within the individual storage cell walls for the purpose of neutron absorption. The racks were supplied with a proprietary material called Boraflex.

As part of the routine surveillance program on the fuel racks, a coupon was removed from the pool for analysis on August 16, 1993. Inspection of the coupon showed approximately 90 percent of the Boraflex material was missing. Three additional coupons were pulled on August 19, 1993. Approximately 38 to 50 percent of the material was found to be missing from these coupons.

The licensee performed preliminary calculations indicating fuel in the pool will remain subcritical without either Boraflex in the racks or soluble boron in the spent fuel pool water, although the five percent negative reactivity (design basis) is not achievable. Consultations are in progress with Westinghouse, INPO, and other nuclear facilities who have Boraflex fuel racks. NRC Information Notice 93-70 was issued regarding this issue and a Part 21 report is anticipated.

e. On August 24, 1993, during performance of an engineered safeguards room cooling and ventilation system test, a safeguards room cooling fan motor failed to start. The licensee found that the contactor for the fan motor had tripped on thermal overload. Upon further review the licensee found that the thermal overload settings for four engineered safeguards fans were reduced prior to the current refueling outage. The reduced settings may not have allowed continuous operation of all fans, causing at least one train of engineered safety features to be inoperable.

   The thermal overload settings for the contactors for all four of the engineered safeguards room cooling fans were increased to their proper setting to ensure the required operability. Followup testing of the fans was satisfactory.

f. On August 30, 1993, the effluent concentration (EC) activity limit for the SIRW tank was exceeded. The limit was exceeded during the transfer of the reactor cavity water back to the SIRW tank. The maximum effluent concentration was 116.8 percent of the tank limits specified in the offsite dose calculation manual (ODCM). The tank was subsequently put on recirculation through the T-50 demineralizer and the activity was reduced to 84.9 percent within the following 24 hours.

g. During heatup from the 1993 refueling outage, a leak occurred in the pressurizer power operated relief valve (PORV) header near its connection to the pressurizer. The licensee identified this leak as a primary coolant system (PCS) pressure boundary leak. The leak was in a non-isolable section of pipe on the PORV line off the pressurizer. The leak appeared to be coming from a 30 degree circumferential crack on a nozzle weld on the pressurizer. The steam plume being emitted from the crack was estimated to be about two feet with a leak rate of approximately 0.25 gallons per minute.

   Upon identification of the PCS pressure boundary leak, the licensee immediately initiated plans to commence a plant cooldown, make repairs, and investigate the root cause. Following return to cold shutdown, inspection found a 3.5 inch crack in the Inconel-600 safe-end of the PORV nozzle. Preliminary metallurgical examination indicated the probable cause to be Primary Water Stress Corrosion Cracking (PWSCC).

   Nondestructive examinations of the nozzle safe-ends and welds during the outage failed to identify the crack even though an indication was noted and evaluated. A region-based metallurgist and specialists from the NRC Office of Nuclear Reactor Regulation were dispatched to the site to evaluate the licensee's activities. This was still in progress at the close of this inspection period.

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

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

Routine facility operating activities were observed as conducted in the plant and from the main control room. Plant startup, steady power operation, plant shutdown, and system lineup and operation were observed as applicable.

The 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.

Periodic verification of Engineered Safety Features status was conducted by the inspectors. Equipment alignment was verified against plant procedures and drawings and detailed walkdowns selectively verified: equipment labeling, the absence of leaks, housekeeping, calibration dates, operability of support systems, breaker and switch alignment, as appropriate.

a. General

The plant completed refueling outage activities and commenced plant heatup on September 15, 1993. The unit entered hot shutdown on September 16, 1993, and remained there for just over one hour when a leak was confirmed from the pressurizer PORV line. The unit was returned to cold shutdown.

The unit exited the inspection period in cold shutdown with repair activities for the pressurizer leak in progress.

b. The licensee completed all remaining actions to initiate a plant heatup, following an extended refueling outage that commenced on June 4, 1993. At 5:50 p.m. on September 16, 1993, the licensee declared the reactor in a hot shutdown condition with reactor temperature greater than 525 degrees F at 2060 psig. During the plant heatup, the licensee identified a leak on top of the pressurizer inside the containment.

The licensee restricted access to the containment until the plant was cooled down below 200 degrees F. Upon the first containment entry to perform a primary coolant system (PCS) walkdown, an auxiliary operator reported that the leak rate had increased. At this time, the operating shift had initiated action to increase the cooldown rate and to depressurize the PCS as soon as possible. Depressurizing the reactor would entail securing the primary coolant pumps (PCP's) and the charging pumps.

The shift supervisor, realizing that securing the PCP's would significantly reduce the heat generated in the primary coolant system and result in a increased cooldown rate, briefed the crew on the cooldown rate restrictions. Technical Specifications (T.S.) 3.1.2.a specifies a maximum heatup/cooldown rate limit of 40 degrees F/hour when the PCS is between 170 and 250 degrees F, and a maximum heatup/cooldown rate of 20 degrees F when the PCS is below 170 degrees F.

As the PCS was greater than 170 degrees F, the operators continued to cooldown at less than 20 degrees F/hour. However, while depressurizing the reactor, the temperature of the PCS decreased below 170 degrees F and the allowable cooldown rate of 20 degrees F/hour was exceeded.

The inspectors consider this excessive cooldown rate to be an unresolved item pending further review for potential enforcement (Unresolved Item No.50-255/93021-01(DRP)).

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

6. <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 available.

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 maintenance activities were observed:

- Insertion/removal of the upper guide structure
- Excore Neutron detector (NE-06) replacement
- Safeguards room cooler thermal overload replacement

- Containment high pressure/containment high radiation relay replacement

The inspectors routinely assessed housekeeping and material condition of the plant. The plant showed overall improvement in both areas throughout the inspection period. Examples of where improvement was noted were the containment on all levels, the auxiliary feedwater (AFW) pump room housing AFW pumps P-8A and P-8B, and the 1C switchgear room. Two areas still needing attention were the service water pump room and the fuel pool heat exchanger room.

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

7. <u>Surveillance</u> (61726, 42700)

The inspector reviewed technical specifications required surveillance testing as described below, and verified that testing was performed in accordance with adequate procedures, test instrumentation was calibrated, and limiting conditions for operation were met. The inspectors further verified that the removal and restoration of the affected components were properly accomplished, test results conformed with technical specifications and procedure requirements, test 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 surveillances were observed:

- Technical Specification Surveillance Test Procedure RO-12, "Containment High Pressure (CHP) and Spray System Tests"
- Special Test Procedure T-339, "Low Pressure Safety Injection System Flow Test (Cold Shutdown)"
- Technical Specification Surveillance Test Procedure QO-30, "Engineered Safeguards Room Cooling and Ventilation System"

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

8. <u>(Closed) Temporary Instruction (TI) 2500/028</u>

The Energy Reorganization Act, Section 211, and 10 CFR 50.7 prohibits employers from discriminating against employees for taking actions to initiate NRC proceedings or otherwise raise safety issues to the NRC or licensees. An issue being assessed by the NRC staff is whether the NRC should encourage or require licensees to implement an employee concerns program to encourage employees to come forward without fear of retribution. To aid in this effort, the inspectors were requested to complete Temporary Instruction (TI) 2500/028, "Employee Concerns Program." The inspector determined that the licensee does not have a formal program to provide employees a means to raise concerns outside of normal line management. A copy of the questionnaire provided with the TI is included as Attachment 1 to this report.

The inspector reviewed Procedure No. 5.23 Rev. 0, "Quality Verification Program," which states that if a verifier cannot achieve adequate resolution to an item through the normal line management, "the item should be forwarded to the Nuclear Performance Assessment Department (NPAD) for assistance of resolution." This procedure applies specifically to conditions identified through the quality verification process, and, therefore, the direction to refer concerns to the NPAD organization is not available to all employees.

Based on interviews with licensee managers, there is an expectation that employees will bring concerns to their line management, and if satisfactory resolution cannot be achieved, that the concern should be raised to the Human Resources Department. The licensee also relies on an "open door" policy which enables employees, who cannot resolve their concerns, to address senior management directly. However, there is no documentation to support any of these policies.

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

9. <u>Management Meeting</u> (30702)

a. A public management meeting was held between D. P. Hoffman, Vice Present Consumers Power Company, and H. J. Miller, Deputy Regional Administrator, RIII, and their respective staffs. The meeting was held at the Holiday Inn, Benton Harbor, Michigan, on September 9, 1993. The purpose of the meeting was to discuss the licensee's progress in addressing the following issues: inadvertent lifting of a fuel assembly with the upper guide structure, damaged fuel assembly I-24, and fuel accountability.

10. <u>Unresolved Items</u>

Unresolved Items are matters about which more information is required in order to ascertain whether they are acceptable items, violations, or deviations. An Unresolved Item disclosed during the inspection is discussed in paragraph 5.b.

11. <u>Persons Contacted</u>

<u>Consumers Power Company</u>

#D. P. Hoffman, Vice President, Nuclear Operations
*#G. B. Slade, Plant General Manager
* R. D. Orosz, Nuclear Engineering & Construction Manager

#R. M. Rice, Director, NPAD
*#T. J. Palmisano, Plant Operations Manager
*#D. W. Rogers, Safety & Licensing Director
*#K. M. Haas, Radiological Services Manager
* J. L. Beer, Radiation Protection Manager
*#R. J. Gerling, Reactor & Safety Analysis Manager
* J. L. Hanson, Operations Superintendent
*#R. B. Kasper, Maintenance Manager
* K. E. Osborne, System Engineering Manager
#R. P. Margol, System Engineering Supervisor
P. Gire, System Engineering Supervisor
#G. H. Goralski, Reactor & Safety Analysis Supervisor
#D. A. Bemis, System Engineer
#W. L. Beckius, Nuclear Performance Assessment Specialist

<u>Nuclear Regulatory Commission (NRC)</u>

#J. G. Partlow, Associate Director for Projects, NRR
#H. J. Miller, Deputy Regional Administrator
* G. E. Grant, Director Designate, Division of Reactor Safety
#W. M. Dean, Acting Director, Project Directorate, III-1, NRR
*#B. L. Jorgensen, Acting Chief, Reactor Projects Branch 2
#A. H. Hsia, Project Manager, NRR
*#R. M. Lerch, Reactor Inspector
B. C. McCabe, Acting Chief Reactor Projects Section 2A
*#M. E. Parker, Senior Resident Inspector
* D. G. Passehl, Resident Inspector

# Denotes those present at the management meeting on September 9, 1993.

* Denotes those present at the exit meeting on September 29, 1993.

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