U. S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

REGION III

Report of Operations Inspection

IE Inspection Report No. 050-155/76-04

Licensee: Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49210

Big Rock Point
Charlevoix, Michigan

License No. DPR-6
Category: C

Type of Licensee: BWR (GE), 240 M t

Type of Inspection: Routine, Announced

Dates of Inspection: February 2-6, 1976

Principal Inspector: D. R. Hunter 3/8/76 (Date)

Accompanying Inspectors: J. E. Kohler

C. M. Erb

Other Accompanying Personnel: None

Reviewed By: E. L. Jordan, Chief
Reactor Projects
Section No. 2
3/11/76 (Date)

8101220238

SUMMARY OF FINDINGS

Inspection Summary

Inspection of February 2-6, (76-04): Review of operations, refueling operations, reportable occurrences, item of noncompliance, plant cleanliness, review and audits, quality assurance, facility tour, and inspector identified and outstanding items.

Enforcement Items

None.

Licensee Action on Previously Identified Enforcement Items

A review of plant modification controls indicates that the licensee actions are not completed. (Paragraph 5g, Report Details III)

Other Significant Findings

A. Systems and Components

Unresolved Item - The inspection results, (radiography and ultrasonic tests) for four welds in the core spray line were lacking third party inspection. This item will be resolved prior to startup. (Paragraph 2, Report Details II)

B. Facility Items (Plans and Procedures)

1. The plant commenced shutdown on January 30, 1976, for the refueling and major modification outage. The outage is scheduled to extend until about May 15, 1976.

2. The licensee had not reviewed the safety aspects of plant personnel regarding fuel handling activities in conjunction with an unlimited number of people inside the containment sphere. A preliminary offsite review was performed during the inspection which indicated no unacceptable personnel risk. (Paragraph 8, Report Details I)

C. Managerial Items

None.

D. Noncompliance Identified and Corrected by Licensee

None.

E. Deviations

None.

F. Status of Previously Reported Unresolved Items

None.

Management Interview

The management interviews were conducted on February 5 and 6, 1976, by Messrs. Kohler, Erb, and Hunter with the following persons present:

C. J. Hartman, Plant Superintendent
R. B. DeWitt, Manager of Production - Nuclear
R. B. Sewell, Nuclear Licensing Administrator
C. E. Axtell, Chemistry and Radiation Protection Supervisor
R. W. Voll, Reactor Engineer
D. E. DeMoor, Technical Superintendent
C. R. Abel, Operations Superintendent
A. C. Sevener, Operating Supervisor
R. E. Schrader, Instrument and Control Supervisor
G. C. Tyson, Maintenance Superintendent
S. E. Martin, Engineer
G. B. Szczotka, Quality Assurance Superintendent

A. The inspector stated that the relatively large number of people in the containment during fueling activities appeared to warrant a review by the plant to determine the adequacy of the containment exits. The licensee acknowledged the statement and indicated that an offsite review had been performed based on the discussion with the inspectors and no unacceptable personnel hazards were revealed. The inspector stated that the item, containment sphere evacuation, would be reviewed in a subsequent inspection. (Paragraph 8, Report Details I)

B. The inspector stated that housekeeping in the containment sphere during refueling was less than desirable particularly with respect to contractor installation of the Reactor Depressurization System. The licensee stated that additional personnel were assigned during the outage and the area of plant housekeeping would be reviewed with the construction forces. (Paragraph 5, Report Details I and Paragraph 6a, Report Details III)

C. The inspector stated that the step ladder and hose connection used to rig the transfer cask with an emergency supply of cooling water in the event of loss of offsite power were not designated as safety equipment and were uncontrolled. The licensee stated that ample time existed to rig the transfer cask with emergency water and therefore these items need not be controlled. The inspector has no further questions regarding this item at this time. (Paragraph 9, Report Details I)

D. The inspector stated that review of the licensee's actions to correct containment instrumentation problems revealed no discrepancies. (Paragraph 2, Report Details III)

E. The inspector stated that the review of operations revealed one item concerning the freezing of the stack gas monitor system for short periods of time during plant operation. The licensee stated that the freezing of the stack gas monitor systems is under review to determine corrective actions.

The licensee also stated that the outage times and air ejector off-gas activity were considered in the calculations of the radioactive releases from the site. The inspector stated that the stack gas monitor freezing was not a new problem at the plant, and the plant should resolve this item expeditiously. The licensee acknowledged the statements made by the inspector. (Paragraph 3d, Report Details III)

F. The inspector stated that a review of refueling operations revealed two items concerning the refueling procedure (RE-02) prerequisites and the testing of the fuel transfer cask safety brake mechanism. The licensee stated that the two areas will be reviewed and appropriate actions taken. (Paragraph 7a, Report Details I and Paragraph 4, Report Details III)

G. The inspector stated that a review of the item concerning the station battery supports will be reviewed further by IE:III and licensing. The licensee acknowledged the statement. (Paragraph 5a, Report Details III)

H. The inspector stated that during the plant tour while exiting the containment vessel personnel hatch, the operating mechanism failed to function properly. The inspector stated that the problem was identified and appropriate corrective actions planned. The licensee acknowledged the statements. (Paragraph 6c, Report Details III)

I. The inspector stated that the failure of the off-gas isolation system to effectively isolate the off-gas stream in the test performed immediately prior to the plant shutdown and the fact that the off-gas isolation system has failed previous tests, indicate a need for an engineering review to determine the unidentified flow path to the stack. The licensee stated that a further review would be provided. (Paragraph 7, Report Details III)

J. The inspector discussed the apparent need of an alternate path for vacuum relief on the containment. The licensee stated that a Consumers Power engineering review was in progress to determine if an alternate path was actually required, and stated that the appropriate actions would be taken. (Paragraph 5c, Report Details III)

K. The inspector stated that during the inspection of the core spray system welds performed in the 1975 outage, questionable practices on specific welds were revealed.

1. The inspector stated that he had reservations about the quality of two welds made during the elevation of the four-inch core spray lines. The authorized inspector had not signed off on the radiography of four welds identified as North, South, 1A and 14A. (Paragraph 2c, Report Details II)

2. The inspector noted that an ultrasonic baseline inspection had been made on the above four welds, with no indication of recordable back reflections from the weld root area, which is extremely irregular as shown by radiograph. (Paragraph 2c, Report Details II)

L. The inspecor stated that an examination of QA documentation for the RDS installation by Catalytic indicated that the work was being processed and accepted to applicable ASME Codes. However, the QA audit by CP did not appear to bear acceptance signatures by CP indicating their approval of material and welds being incorporated into the RDS. (Paragraph 3, Report Details II)

M. The inspector noted the use of clear poly around and near the spent fuel pool and the reactor vessel opening was considered questionable considering the difficulty in locating poly under water. The inspector asked the licensee to review the use of such materials. The licensee stated that the question of using clear poly would be reviewed. (Paragraph 5b, Report Details III)

REPORT DETAILS

Prepared By: J. E. Kohler 3/8/76
J. E. Kohler (Date)

Reviewed By: W. S. Little 3/9/76
W. S. Little (Date)

1. Persons Contacted

C. J. Hartman, Plant Superintendent
R. Voll, Reactor Engineer
J. L. Keumin, Maintenance Engineer
C. E. Axtell, Chemistry and Radiation Protection Supervisor

2. Pre-Refueling Activities

The inspector verified that surveillance testing involving the prerefueling activities have been completed.

a. Preparation of the transfer cask for refueling (procedure MFHS-1).

b. Crane testing - testing of the fuel handling cables (procedure MFHS-2).

c. Refueling interlock test (procedure TR-02).

d. Communication systems verification.

e. Cooling capability for stored fuel.

3. Refueling Deck Radiation Monitors

Radiation monitoring on the refueling deck available for protection of the refueling crew consisted of a Continuous Air Monitor (CAM). The monitor was checked for operability daily on day shift by verifying that there is flow through the monitor and that the strip chart was functioning properly.

4. Fuel Handling Activities

The inspector verified by record review and direct observation that the following conditions existed and fuel handling activities during refueling were being conducted according to approved procedures.

a. Core monitoring consisting of two source range monitors were reading greater than two counts per second.

b. Containment integrity was maintained consisting of the double door air locks being closed and the containment sphere vent valves open during fuel movement.

c. The inspector verified that insertion and removal of fuel bundles was in accordance with approved fuel bundle removal procedure RE-02.

d. The inspector verified that fuel accountability, consisting of a bundle unloading sequence specifying spent fuel pit locations, as well as tagging procedures in the control room, was in accordance with approved procedures.

e. Core internals were protected with polyethylene.

f. The make-up of the refueling crew on the deck and in the control room was in accordance with the established plant procedures. The minimum crew requirement consisted of a roving shift supervisor, one licensed operator in the control room, two licensed operators on the reactor deck and an auxiliary crane operator.

g. Water level and water temperature in both the spent fuel pit and the reactor were being monitored by control room operators and auxiliary plant operators.

h. The reactor mode switch was in the refueling mode.

i. The inspector verified that a licensed operator was present in the control room and in constant communication with the fuel handling crew during all fuel movement.

5. Housekeeping

Housekeeping inside the containment sphere during fuel movement was poor, particularly with respect to contractor personnel inside the reactor sphere. The inspector noted unsecured power cables, welding bottles and litter in the sphere.

6. Previously Reported Unresolved Item

The inspector reviewed the unresolved item[1/] pertaining to the fire stop penetration sealant material qualification. The fire stop modifications, indicating the unresolved item, is still in engineering and the fire stop field activities have not commenced. This item remains open.

1/ IE:III Inspection Report No. 050-155/75-16.

7. Previously Reported Open Items

a. Revisions to Refueling Procedure RE-02

1. The inspector determined that fuel handling procedure RE-02 revisions included a bundle unloading sequence specifying spent fuel pit location, and the prerequisite section included the appropriate refueling procedures requiring signoff prior to any fuel movement.[2/]

2. The fuel handling procedure (RE-02) prerequisites did not contain all the required systems and components required by the Technical Specifications during refueling operations. It was determined through procedure review and discussions with the licensee that the refueling procedure (RE-02) should cover the plant conditions and requirements since the master checklist is not effective during this period of time after plant shutdown until plant refueling. The inspector reviewed selected systems required during the refueling operations by the Technical Specifications and noted no discrepancies.

b. Ventilation Requirements in Fuel Storage Areas

The licensee is handling fuel with the containment vent valves open. Upon high radiation signal, the refueling crew will notify the control room operator by telephone to isolate the containment sphere. NRR is aware of this procedure and is currently reviewing the necessity of automatic containment isolation on high radiation for installation prior to the next refueling outage (cycle 14). The inspector considers this item closed out.[3/]

c. Refueling Radiation Deck Monitors

The CAMS do not annunciate in control room, however, two area monitors required for monitoring criticality annunciate in the control room. The licensee stated that any significant fuel handling incident would cause the control room area monitors to annunciate. As stated above, subsequent control room operator action would be to close the containment sphere ventilation isolation valves. The inspector considers this item closed out.[4/]

2/ IE:III Inspection Report No. 050-155/76-01.
3/ Ibid.
4/ Ibid.

d. Fuel Sipping

The procedure for dry sipping has not been completed by the licensee. This item remains open and will be followed at the next inspection.

e. Fuel Inspection

The procedure for vendor fuel inspection and bundle reconstitution of reactor fuel has not been approved by the licensee. This item remains open and will be followed at the next inspection.

8. Emergency Evacuation of the Containment Sphere

The licensee is handling fuel in parallel with the installation work on the reactor depressurization system (RDS). The RDS installation and fuel handling activities can involve thirty or more people in the sphere at any one time. Because of the large number of people that could be in the sphere during the fuel movements, the inspector discussed provisions for timely evacuation of the sphere in the event of an emergency, giving consideration to the capacity of the air lock and the length of time to pass through the air lock. The licensee completed a preliminary review of this item while the inspectors were at the site. The licensee's preliminary review indicated no unacceptable personnel hazards would result from a postulated fuel handling accident.

9. Emergency Cooling for Fuel Transfer Cask

The inspector noted that the step ladder and hose connection used to rig the transfer cask with an emergency supply of cooling water were not designated as safety equipment. As such they are uncontrolled and cannot be assumed to be available when needed. The licensee stated that these items need not be controlled because the FHSR calculated 200 minutes time period before the water in fuel cask boiled away. The inspector has no further questions regarding this item.

REPORT DETAILS

Part II

Prepared By: C. M. Erb 3/5/76
C. M. Erb (Date)

Reviewed By: J. C. LeDoux 3/8/76
J. C. LeDoux (Date)

1. Persons Contacted

The following individuals were contacted during the inspection.

Consumers Power Company (CP)

G. C. Tyson, Maintenance Superintendent
S. E. Martin, Engineer - Maintenance
R. Stafford, Inspector - Radiography
H. Keiser, Engineer - Operations

Catalytic Construction Company (Catalytic)

J. Chapman, Supervisor - Quality Assurance
G. Kenny, Quality Control Inspector

2. Core Spray Valve Relocation

a. Reason for Change

Two isolation valves, a check valve, and associated piping were raised several feet in elevation, so that malfunction of the isolation valves, due to flooding, could not occur. This involved removing about 17 feet of pipe and then re-welding the system.

b. Materials and Specifications

The Powell isolation valves are carbon steel, No. MO-7051 and No. MO-7061, and the connecting piping is four-inch diameter to Specification ASTM A-106, Grade B. Wall thickness of part of the pipe is a nominal .237". Procedure No. MPIS-5, Revision 6, covers relocation of two valves and one flow element. The piping was installed to the requirements of ANSI B31.1 - 1973, Power Piping Code. The radi-graphic and penetrant inspection acceptance were to be based on Section III, NB5300, while repair was based on Section XI, IWB4423. In general, the rerouting was performed to ASME Section XI, 1971 edition, paragraph IS-400.

c. Qualit Assurance Results

Records for four welds, No. 1A, No. 14A, No. N, and No. S, were examined. These welds were made using a V-groove preparation. The gas tungsten arc process was used with an open butt to fuse in the root. The welds were completed using the shielded metal arc process. Procedure and personnel qualifications to Section IX were in the file.

The radiographs indicated many repairs had been made with porosity, due to loss of protective gas and burnthrough which resulted in thin and thick areas of weld. Welds No. South and No. 14A showed the poorest quality. An identification of two welds as North and South in the system could lead to problems, and it is recommended that welds should have a number or letter in the weld number which identifies the system. A number band, which located areas around the circumference of the weld, was used, but the paper work for repair did not indicate required grinding areas for weld repair. A baseline UT inspection was performed on the above welds, but no indication of root abnormalities was shown.

The prerequisites for the activity were signed off by the code inspection but the final repair including review of radiographs and ultrasonic testing was not signed off.

3. Reactor Depressurization System (RDS)

a. An automatic depressurization system for the reactor system is being installed at the present time. Catalytic Construction Company (Catalytic) has the contract to install the headers, valves, and piping for this system. Grinnell Company (Grinnell) has furnished shop-welded spools for this job. Four isolation valves were supplied by Anchor-Darling Company, and four safety relief valves were supplied by Target Rock.

b. Quality Assurance

The inspector examined the radiographs and other NDE records for the following welds and found them acceptable to QAP-7125.

Weld	Origin	Fabricator	Size	Process	Welder
A-RDS-101-J1N1-5	Shop	Grinnell	12"	Automatic	-
B	Shop	Grinnell	12"	Automatic	-
102-D-1	Field	Catalytic	6"	Manual	PF-9
101-6	Field	Catalytic	12"	Manual	PF-4

Catalytic is using the gas tungsten arc process with a Grinnell insert for the weld root, followed by shielded metal arc process to completion. Section III, 1974, edition of the ASME Code, is the governing document. The Catalytic procedures and personnel were qualified to ASME Section IX.

The Class 1 valves were produced with an "N" Stamp affixed, and certifications as to materials were in the Catalytic QA files. Certifications, as to minimum wall thickness, were also in the file and satisfactory.

The inspector understood that CP quality assurance representatives are auditing the QA results of the Catalytic operation. However, no signatures of CP QA representatives for acceptance were seen on the quality documentation.

4. In-service Inspection

a. Status Inspection

CP has contracted the in-service inspection to Southwest Research Institute (SWRI) for the reactor vessel and recirculation piping. CP expects to perform examination of the steam drum and supports. U-Tech Company are also expected to do some of the inspection work.

SWRI will make as-built isometrics of the piping systems with particular emphasis on locating and inspecting all accessible bimetallic welds. SWRI or CP will determine the length of longitudinal welds in vessels requiring a percentage inspection over a ten-year period.

Eight Class 1 valves are scheduled for valve wall thickness determination during this outage.

The inspector was shown an overall inspection plan indicating the number of welds which will be updated when SWRI completes its work. No procedures from SWRI had been approved, and a meeting was held on February 5, 1976, at CP corporate headquarters with the authorized inspector, at which time all procedures and NDE plans were to be approved. SWRI was scheduled to begin work on February 10, 1976.

REPORT DETAILS

Part III

Prepared By: D. R. Hunter

1. Persons Contacted

   C. J. Hartman, Plant Superintendent
   D. E. DeMoor, Technical Superintendent
   C. R. Abel, Operations Superintendent
   G. C. Tyson, Maintenance Superintendent
   G. B. Sczczotka, Quality Assurance Superintendent
   C. E. Axtell, Chemistry and Radiation Protection Supervisor
   A. C. Sevener, Shift Supervisor
   R. E. Schrader, Instrument and Control Supervisor
   S. G. Martin, Plant Engineer
   F. J. Valade, Shift Supervisor
   T. M. Brun, Assistant Chemistry and Radiation Protection Supervisor
   S. A. Carlisle, Shift Supervisor
   R. W. Doan, Training Coordinator, Shift Supervisor
   C. F. Sonnenberg, Assistant Shift Supervisor
   J. J. Zabritski, Quality Assurance Engineer

2. Review of Reportable Occurrence Reports

   a. P-01-76, inadequate design pressure ratings on a vacuum transmitter and six pressure switches, reported on January 19, 1976. The licensee reported[5/] that during an evaluation of the containment penetrations regarding Appendix J to 10 CFR 50, the vacuum transmitter PT-173; containment pressure switches, PS-664-667; and containment pressure/vacuum switches, DPS-9051 and DPS-9052, would not withstand the containment pressure of 23 psig anticipated during the DBA. The inspector reviewed operating memo 2-76, logged in the Shift Supervisor's log on January 21, 1976, which was in effect until the end of cycle 13. The containment vacuum transmitter and the containment pressure/vacuum switches were noted as isolated and were required by the operating memo to be unisolated at a containment pressure below five psig following a containment pressurization event. The inspector verified that the licensee plans to replace the switches during the present outage.

5/ CP to IE:III, ltr dtd 2/2/76.

3. Review of Plant Operations

The inspector reviewed the following selected records of routine plant operations to verify these activities to be in accordance with the Technical Specifications and Administrative procedures.

a. Shift Supervisor Log, January 8, 1976 through February 2, 1976.

b. Control Room Operator Log, January 13, 1976 through February 1, 1976.

c. Reactor Operator Log, January 6, 1976 through February 4, 1976.

d. Control Room Data Sheets, January 1, 1976 through February 3, 1976.

(1) The inspector noted that the stack gas monitor had been logged as frozen and no readings taken on several occassions during January.

(a) 1-18-76, 1400-2400, 10 hours

(b) 1-19-76, 00-0800, 8 hours

(c) 1-22-76, 1300-1600, 3 hours

(d) 1-23-76, 1300-1500, 2 hours

(e) 1-24-76, approximately 6 hours

The stack gas monitor, including the iodine and particulate filters were out of service in January for approximately 29 hours, and a total of 18 hours continuously on January 18-19, 1976. The inspector verified that a QA-16 had been issued (November 25, 1975), and that the problem was being pursued to correct the apparent freezing of moisture in the embedded lines in the stack between the isokinetic probe and the equipment at the base of the stack. The inspector verified through discussions with the licensee representative that the filter flows and any activity changes during January were considered during the outage times.

e. Operating Memos.

f. Daily Orders, January 6, 1976 through February 2, 1976.

g. Control Room Status Board.

h. Outstanding tags and tagging orders.

i. Fuel Status Boards.

j. Administrative Key Control.

4. Review of Refueling Operations

a. The inspector reviewed the master checklist and selected systems checklists to determine that systems disturbed during the refueling outage will be returned to normal prior to unit refueling and plant startup operations. No discrepancies were noted.

b. The inspector's review of activities associated with the fuel transfer cask revealed that the safety cable braking mechanism was not addressed relative to inspection or testing requirements; nor was the inspection or testing of the mechanism included as a scheduled preventative maintenance item. The licensee's correspondence[6/] with licensing indicated that the braking mechanism was the basis for moving the fuel transfer cask directly from the core to the spent fuel pit at an elevation of approximately 1½ feet above the refueling deck. This elevation above the floor allowed the 10 inch distance required to activate the safety braking mechanism in the case of a fuel transfer cask drop accident. The inspector verified that the safety braking mechanism had been accidentally tripped during the cask rigging operations immediately prior to fuel movements. The licensee indicated that the mechanism had been tripped on several other occasions during the past. This item will remain open pending the completion of a review by the licensee to determine the inspection and testing requirements of the safety braking mechanism.

5. Review of Outstanding Items

The inspector reviewed selected outstanding items to determine licensee followup actions.

a. The station battery seismic requirements were addressed in Consumer's Power internal correspondence dated December 19, 1975, in answer to AIR BRP 67-75, which requested an engineering evaluation. The evaluation indicated that the battery was installed in accordance with the Final

6/ CP to NRR, ltr dtd 1/22/76.

Hazards Summary Report, section 2.6, and additional support for the battery was not required. The FHSR indicates the containment, concrete structure, and the equipment are designed to at least maximum ground acceleration rate of 0.05 gravity. The containment vessel is designed to withstand a wind force on the vessel of 100 miles per hour, which exceeds the earthquake forces. The equipment installed outside the containment vessel does not appear to be addressed in the FHSR. The station battery is the power supply for the DC-ECCS valves (core spray and building spray valves) and the power supply for various other engineered and operational safety features, including:

(1) Liquid poison system controls.

(2) Reactor building ventilation and vacuum breaker valves.

(3) 480V motor control center 2B control power.

(4) Protection Bus #3.

(5) 2400V switchgear control power.

(6) Emergency Condenser outlet valves.

(7) Control room annunciators.

This item will remain open pending further review by IE:III.

b. The inspector reviewed the off-gas isolation test performed on January 31, 1976, (O-WGS-1). The off-gas holdup line isolated and the pressure increased as expected over the duration of the test. The stack gas activity initially decreased, but immediately increased again; indicating an unidentified off-gas isolation system bypass flow path. The condenser vacuum did not change substantially throughout the test period, indicating that a bypass flowpath exists. The inspector reviewed the off-gas isolation with the licensee, which has been an outstanding item since June 1972.7/ This item will be followed during a subsequent inspection trip.

c. The inspector reviewed the communications between the licensee and NRR concerning the containment vacuum relief system. The inspector noted that the controls for manual operation of the vacuum breaker valves and an alternate flow path were being reviewed by the licensee. This item will remain open pending completion of the review by the licensee.

7/ CP to DL, ltr dtd 6/26/72.

d. The inspector reviewed the failure analysis package associated with the control rod drive relief valve nipple failure.[8/] The failure was due to fatigue caused by vibrations associated with the positive displacement charging pumps. The licensee has initiated an engineering review to provide supports for the CRD piping subject to failure by vibrations during CRD pump operation. No discrepancies were noted.

e. The inspector reviewed the revised administrative procedures concerning temporary and permanent procedure changes to operating procedures (1.4.A.6) and to operations checklists (1.4.A.3.5.1).[9/] No discrepancies were noted.

f. The inspector reviewed with the operations superintendent the corrective actions concerning review of the maintenance procedures associated with the control rod drives as a result of AO 050-155/25-75. The licensee representative indicated that during 1976, procedures for major maintenance items are planned to be written.

g. The inspector reviewed the corrective actions associated with the item of noncompliance.[10/][11/] This item remains open.

   (1) The required circuit analysis was not completed.

   (2) The plant review committee is continuing to determine each facility change as safety related or not immediately prior to commencing the facility work until item (4) below is completed.

   The inspector reviewed selected construction activities management controls to insure the activities (work packages) safety reviews were being performed by the plant review committee prior to commencing the specific work activities.

      (a) Selected work packages indicated completion of safety evaluations of the plant interfaces.

      (b) Field change notices were reviewed by the responsible engineer and if necessary, due to being outside the original safety evaluation, were approved by the plant review committee.

8/ AO 050-155/29-75.
9/ IE:III Inspection Report No. 050-155/75-10.
10/ IE:III Inspection Report No. 050-155/75-15.
11/ CP to IE:III, ltr dtd 12/19/75.

(c) The selected construction activities were being performed within the approved work packages.

(3) The Q-list is in effect.

(4) The corporate and plant procedures relating to modification control have not been revised.

6. Facility

The inspector toured the facility to view the plant and construction activities.

a. The construction areas were becoming cluttered with work items such as hoses, extension cords, and miscellaneous items. The inspector noted that the licensee had extra personnel assigned to remove trash from the construction areas.

b. The inspector noted the use of clear polyethylene sheeting for general purposes around the spent fuel pit and the open reactor vessel. The inspector questioned this practice based on the difficulty of seeing and retrieving clear poly or clear plastic from the pit or the reactor vessel and the possibility of returning the plant to operation with plastic items in the primary coolant system.

c. The inspector reviewed the operation of the containment personnel hatch electro-hydraulic operating mechanism following the improper operation of the interlock system on the inner door during containment exiting. The malfunction, O-ring compression, was apparently caused by the increased usage of the doors during the major construction activities. The inspector verified that the surveillance of the access hatch hydraulic pressure will be increased pending a further review by the licensee and a possible permanent change to the door operating mechanism.

d. The inspector reviewed the plant illuminated annunciators with a control room operator. No discrepancies were noted.

7. Quality Assurance

The inspector reviewed with the licensee representative the present status of the implementation of the quality assurance program.

a. The Consumers Power quality assurance policies are in final review in Licensing at the present time.

b. The licensee plans to establish implementation dates associated with the specific areas of the quality assurance program manual for nuclear power plants.

c. A selective review of the quality assurance program procedures indicated that the licensee is presently involved in implementing program requirements.

d. The plant quality assurance group appears to be performing plant audits and surveillance on a limited basis. An audit plan has been issued but is not yet approved.