U. S. ATOMIC ENERGY COMMISSION
DIRECTORATE OF REGULATORY OPERATION

REGION III

Report of Operations Inspections

RO Inspection Report No. 050-010/74-06
RO Inspection Report No. 050-249/74-05

Licensee: Commonwealth Edison Company
P.O. Box 767
Chicago, Illinois 60690

Dresden Nuclear Power Station
Units 1 and 3
Morris, Illinois

Licenses No. DPR-2 and
No. DPR-25
Categories: C

Type of Licensee: G.E., BWR, 200 Mwe and 800 Mwe (Net)

Type of Inspection: Routine, Unannounced

Dates of Inspection: May 7-10, 16, 17, 28-31 and June 7, 1974

Dates of Previous Inspection: May 6-8, 10, 15, 1974 (Materials)

Principal Inspector: F. Maura F Maura 7/25/74 (Date)

Accompanying Inspector: None

Other Accompanying Personnel: None

Reviewed By: H. Dance H C Dance 7/26/74 (Date)
Senior Reactor Inspector

8009030716

SUMMARY OF FINDINGS

Enforcement Action

A. Violations considered to be of Category II severity are:

1. Units 1 and 3 Technical Specification 6.2.A(7) requires that detailed written procedures including applicable check off lists covering surveillance requirements shall be adhered to:

   Contrary to the above:

   a. The data required under Step #9 of the Unit 1 battery bank discharge test was not taken. (Paragraph 6.a)

   b. During the performance of local leak rate tests on Unit 3, the air supply line to the penetrations was not disconnected as required by procedure 38-1600-S-0 Step 8 and in the case of the "B" MSIV line by procedure 38-3000-S-I Step F. (Paragraph 7.d)

2. Technical Specification 6.2.D requires that work instructions or special test procedures for the maintenance staff require compliance with Radiation Control Procedures.

   Contrary to the above, two maintenance personnel were observed violating Radiation Control Procedures on May 16, 1974, during the LLRT of the "B" MSIV of Unit 3. (Paragraph 7.d)

3. Unit 1 Technical Specification 6.2.A.(2) requires that detailed written procedures including applicable check off lists covering refueling operations shall be prepared, approved and adhered to.

   Contrary to the above, the removal of the orifice plate from Unit 1 core position 55-08 with a stainless steel J-hook was performed without an approved written procedure. As a result excessive force was applied which resulted in the loss of the J-hook in the reactor vessel. (Paragraph 5.b)

4. 10 CFR Part 50, Appendix B, Criterion XVII requires that sufficient records be maintained to furnish evidence of activities affecting safety. The records shall include qualification of equipment.

   Contrary to the above, maintenance performed on the Unit 1 "B" core spray pump breaker on December 27, 1973, required the installation of a new trip unit. Documentation to show that a like-for-like replacement was installed was not available to the inspector. (Paragraph 8.a)

5. 10 CFR Part 50, Appendix B, Criterion V, states in part that, "Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions."

   Contrary to the above, the modification to the Unit 1 standby liquid control system was performed, tested and signed as completed without the required M&S Engineering approval to deviate from the requirement of USAS B 31.1 paragraph 137.1 and the test requirements stated in the modification package as required by Quality Assurance Procedure Q.P. No. 3-51, Rev. 2, Step 17. (Paragraph 9.a(1)).

6. 10 CFR Part 50, Appendix J requires that all Type A tests shall be conducted in accordance with the provisions of ANSI N45.4-1972.

   By letter to the AEC dated September 26, 1973 (J. Able to D. Ziemann) Commonwealth Edison Company committed to use ANSI N45.4-1972 data reduction methods.

   Contrary to the above:

   a. Data reduction of temperature measurement was not done in accordance with paragraph 7.4 of ANSI N45.4-1972.

   b. A dated log of events was not properly maintained in accordance with paragraph 7.8 of ANSI N45.4-1972.

   c. Computation of the leakage rate throughout the test was not done in accordance with paragraph 7.9 of ANSI N45.4-1972. (Paragraph 7.e)

7. Unit 3 Technical Specifications 4.7.A.2.a(2) states that leak rate tests shall be performed without preliminary leak detection surveys on leak repairs immediately prior to or during the test. Paragraph (3) states that leak repairs if necessary shall be preceeded by local leak rate measurements where possible. The leak rate difference prior to and after repair when connected shall be added to the final integrated leak rate result.

   10 CFR Part 50, Appendix J, Section III, requires that the Type A test be made so that the containment can be tested in as close to the "as is" condition as practical.

   Contrary to the above, the Unit 3 primary containment "as found" condition can not be actually determined because the licensee failed to test five double gasketed penetrations prior to their opening at the beginning of the outage. (Paragraph 7.d)

8. 10 CFR Part 50, Appendix B, Criterion XII requires that instruments and other testing devices used in activities affecting quality be properly controlled, calibrated and adjusted at specified periods to maintain accuracy within necessary limits. Criterion XVII requires that sufficient records be maintained to furnish evidence of activities affecting safety. The records shall include qualification of equipment.

   Contrary to the above, calibration records for several of the test instruments used during the performance of the ILRT on Unit 3 did not exist. (Paragraph 7.e(3)).

Licensee Action on Previously Identified Enforcement Action:

The corrective actions listed in the licensee's response to our letter of February 28, 1974, were reviewed. The items have been completed although one was 2 months after the date stated in the letter. (Paragraph 2)

Unusual Occurrences

A. Scram outlet valve failed to open. (Paragraph 4.c(4)).

B. CRD's B-2 and J-3 failed to withdraw. (Paragraph 4.c(2) and (3)).

C. CRD F-1 jammed by cap screw. (Paragraph 4.c(1)).

D. Stainless steel J-hook is lost in reactor vessel. (Paragraph 5.b)

E. "B" core spray pump breaker failure. (Paragraph 8.a)

F. Abnormal degradation of "C" core spray pump minimum flow line. (Paragraph 8.b)

G. Standby liquid control "B" pump failure. (Paragraph 9.a(3)).

Other Significant Findings

A. Current Findings

1. The following organizational changes have been made at Dresden since our last routine inspection on March 1974 (Units 2 and 3):

   a. Effective May 15, 1974, E. Meintel replaced O. Dodd as Maintenance Engineer.

b. Effective June 10, 1974:

E. Budzechowski replaced G. Abrell as Operating Engineer for Unit 1

G. Abrell replaced R. Ragan as Operating Engineer for Unit 2

R. Ragan filled the new position of Operating Engineer for Rad Waste and Major Modifications

N. Jackiw replaced E. Budzechowski as Quality Assurance Engineer

M. Wright replaced N. Jackiw as Quality Control Lead Engineer

J. Lamping filled the new position of Staff Assistant-Mechanical to the Maintenance Engineer

J. Jurecki filled the new position of Staff Assistant - Electrical to the Maintenance Engineer.

2. Unit 1 continues in a refueling maintenance outage which began last October. Startup was expected by end of June 1974. Unit 3 startup is planned for first week in June.

B. Status of Previously Reported Unresolved Items: None

Management Interview

The following subjects were discussed on June 7, 1974, with Messrs. B. Stephenson, Station Superintendent; A. Roberts, Assistant Station Superintendent; J. Diederich, Administrative Assistant to Station Superintendent; N. Jackiw, new Q.A. Engineer; and E. Budzechowski, outgoing Q.A. Engineer, and on June 25, 1974 with Mr. B. Stephenson.

A. The inspector read the list of violations to the Technical Specification or AEC regulations which had been identified during the inspection (Enforcement Action).

B. The inspector stated that he does not consider the licensee's performance of the LLRT and ILRT for Unit 3 much better than past performances such as Unit 1 ILRT last January. Specific examples given were:

1. Errors in data taking due to personnel unfamiliarity with the equipment or carelessness.

2. Poor log of events and keeping test personnel on shift informed as noted by one shift averaging 19 temperature points while other two shifts were using 20 points and the fact that the log could not be used to learn what leaks had been fixed, when or how.

3. Poor planning as it was noted that just before the test is to commence the procedure had to be changed several times and the methods to calculate vapor pressure clarified. Also the fact that test did not conform to ANSI N45.4-1972.

4. Temporary procedure changes were being processed in violation of T/S requirements until brought to the attention of the licensee by the inspector.

5. Personnel performing the test were inexperienced (not one had ever been involved in an ILRT before) and appear to have received very little, if any, guidance from station management.

6. Procedure not followed during conduct of LLRT's.

7. Five double-gasketed penetrations not tested in the "as found" condition.
(Paragraphs 7.d and 7.e)

C. The inspector asked what plans the licensee had for the repair of the two major leaks found during the ILRT. The licensee stated the communications penetration would remain sealed until plans for its repair and retest could be developed. With respect to the drywell cooler damper control instrument air lines the valves will be left closed. The Unit 2 valves will be closed also.

D. The inspector noted that two additional sampling points had been installed in the Unit 1 sphere East and West pipeways. Also the resolution of the condition of Unit 1 fuel element UN-350. Both items are considered resolved. (Paragraphs 3 and 5.a)

E. The inspector noted the lack of a report from General Electric on the metallurgical condition of control rod blade B-87 removed from the core in late 1971. The inspector suggested management followup to obtain the report as soon as possible.

F. The inspector discussed the findings of his inspection of the torus-to-drywell vacuum breaker position indicating switches and stated that if problems with the system continue the licensee must inform DL that the present system connot reliably determine when a valve is open 1/8" and proposes some alternate resolution.

The licensee stated he thinks a solution to the counter weight arm-to-valve shaft slipping problem has been found. (Paragraph 7.b)

G. The inspector stated their actions to the letter of noncompliance dated February 28, 1974, were reviewed and there are no further questions at this time. (Paragraph 2.b)

H. The inspector discussed licensee's commitments which had not been carried out satisfactorily. The licensee felt their new systems have started closing the cracks and that by mid summer this problem will not exist any longer. (Paragraph 2.b)

REPORT DETAILS

1. Personnel Contacted

B. Stephenson, Station Superintendent
J. Diederich, Administrative Assistant to Station Superintendent
T. Watts, Supervisory Engineer, Technical Staff
G. Abrell, Operating Engineer Unit 1
D. Scott, Operating Engineer Unit 3
E. Meintel, Maintenance Engineer
D. Adam, Radiation Control Engineer
M. Turbak, Engineer, Technical Staff
R. Bishop, Engineer, Technical Staff
R. Canalas, Engineer, Technical Staff
J. Bowers, Engineer, Technical Staff
T. Lang, Engineer, Technical Staff
J. Wujciga, Engineer, Technical Staff
J. Sierzant, Engineer, Technical Staff
J. Dolter, Engineer, Technical Staff
R. Herbert, Engineer, Technical Staff
N. Jackiw, Quality Control Engineer, Technical Staff
M. Wright, Quality Control Engineer, Technical Staff
E. Meadows, Engineering Assistant, Technical Staff
R. Cozzi, Engineering Assistant, Technical Staff
G. Zwarich, Engineering Assistant, Technical Staff
R. Dyer, Maintenance Foreman
R. Jeffers, Engineering Assistant, Maintenance
B. Zank, Engineering Assistant, Training
J. Rivello, Chief, Engineer, Long Island Lighting Co. on Special Assignment at Dresden Technical Staff

2. Organization and Administration

a. Record Review

(1) Deviation Reports 12-1-74-1 to 12-1-74-7

(2) Incident Reports 12-1-74-1 and 2

(3) Shift Engineers Log - May 5 - 15, 1974

(4) Unit 1 Operators Log - February 14 - 22, 1974

b. Licensee's Commitments to AEC

(1) Response to Items of Noncompliance - Unit 1

The corrective measure to items of noncompliance outline in the licensee's letter (Lee to Keppler) of March 20, 1974 were

inspected. All items have been completed satisfactorily, although it should be noted that the last item, the revision to the emergency procedures, was completed approximately two months after the date of the letter.

(2) Procedure to Prevent Inadvertent Criticality - Units 1, 2 and 3.

The commitments outlined in the licensee's letter (Lee to Keppler) dated January 21, 1974 were reviewed. Open items remain as follows:

(a) Paragraph 1.a(2) is in progress. Final completion March 1, 1975.

(b) Paragraph 1.b.: The licensee has done nothing although the letter states completion was due March 21, 1974. The last revision to the jumper procedure was on November 1973.

(c) Paragraph 1.c.: Procedure change for Unit 1 to be completed before next refueling. For Units 2/3 the licensee proceeded to combine all the rules, precautions, etc. into a single procedure (800-VII issued March 1974) written in the form of a letter. The inspector suggested to the licensee that each precaution, limitation, rule, etc. should be a part of each specific procedure to which it is applicable.

(3) Inspection Commitment

During the April 1973 inspection[1]/ the licensee committed to write a procedure to cover the loss of all d.c. power on Unit 1. During the same inspection, the licensee was reminded of a 1972 commitment to develop a procedure including action points at which the unit would be shutdown following the loss of the battery chargers. As of the date of this inspection the licensee has not started on either commitment.

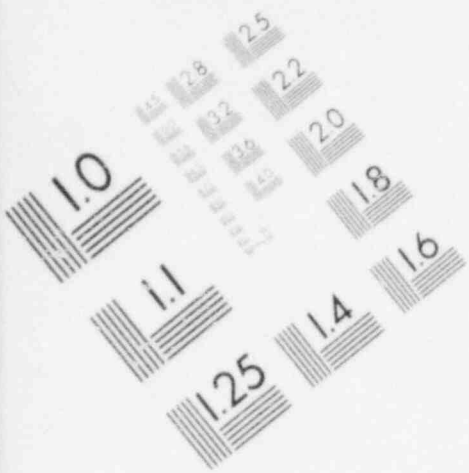
It should be noted that the above also fails the commitment given by the licensee to DL by letter dated March 1, 1973 that it would comply with Safety Guide 33 by March 1, 1974.

3. Reactor Coolant

a. Leakage Sampling - Unit 1

Two additional sampling points were installed during the outage, at elevation 517 feet, to more efficiently monitor the pipeways inside

1/ RO Inspection Rpt No. 050-010/71-01.



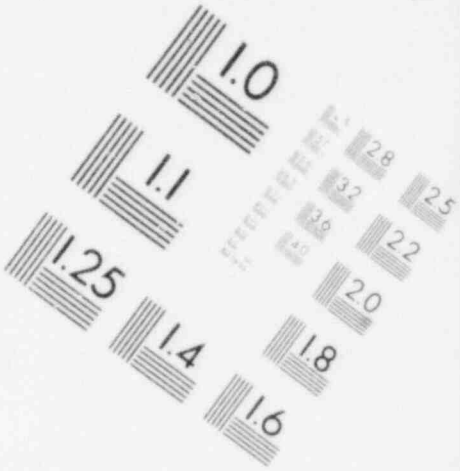
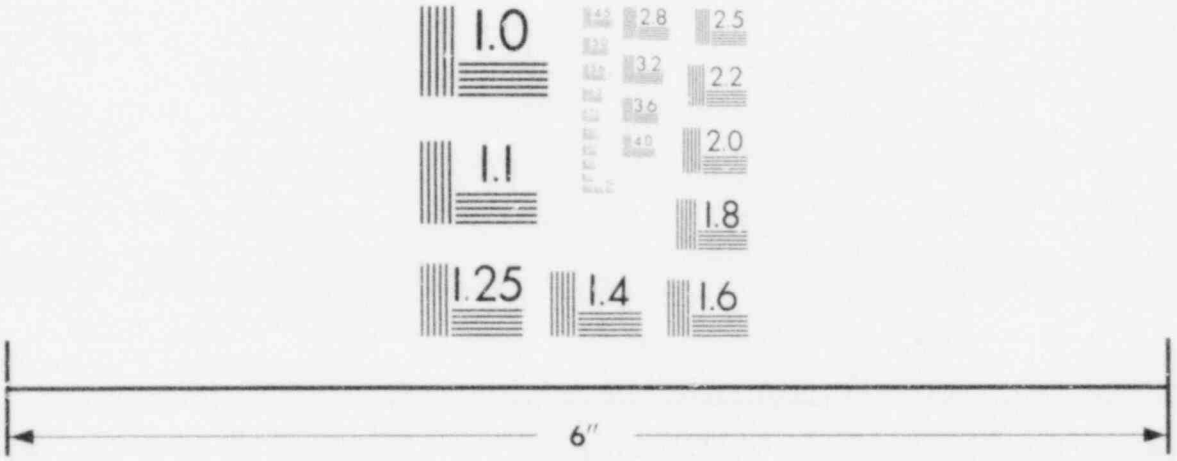
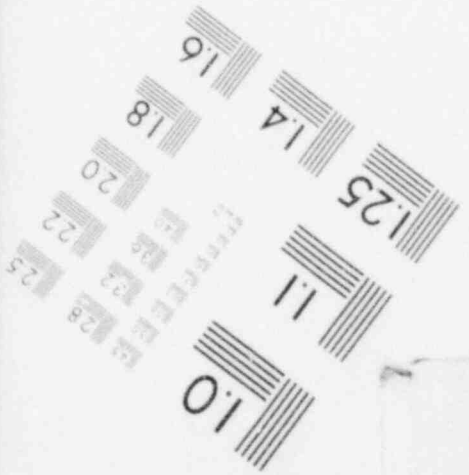
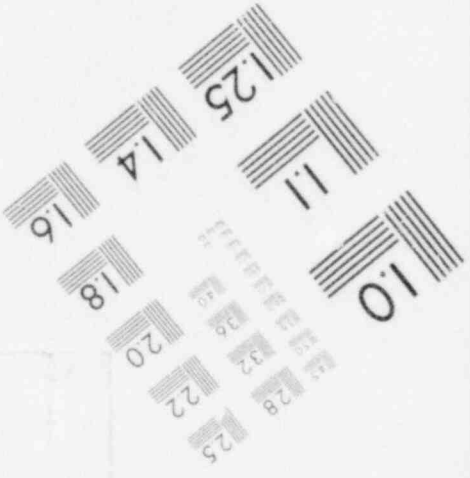


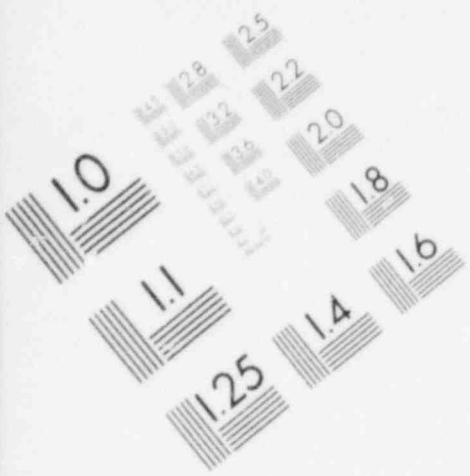
IMAGE EVALUATION
TEST TARGET (MT-3)



MICROCOPY RESOLUTION TEST CHART







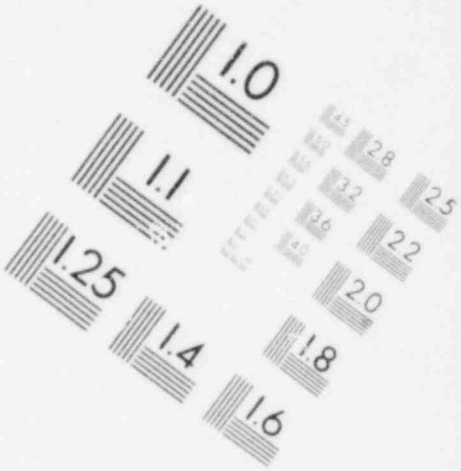
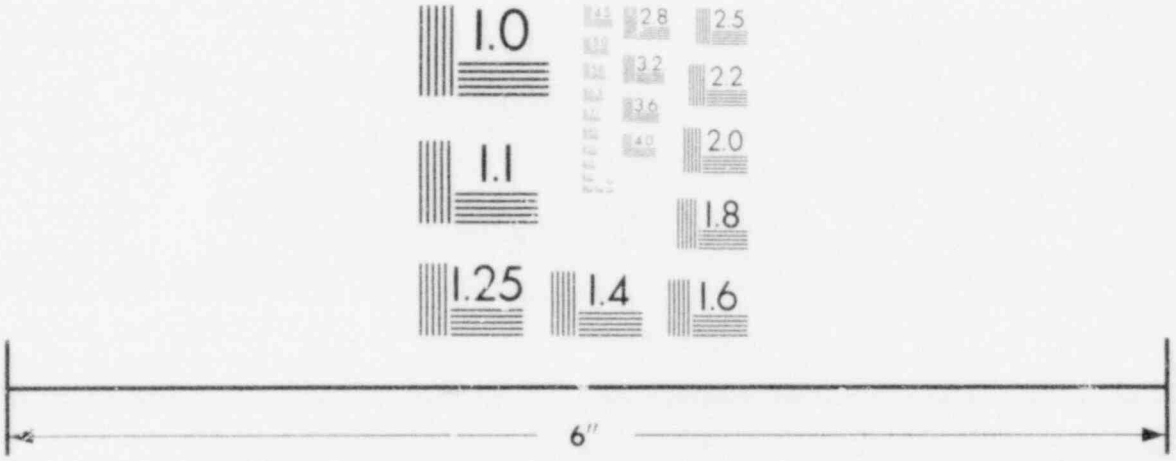


IMAGE EVALUATION
TEST TARGET (MT-3)



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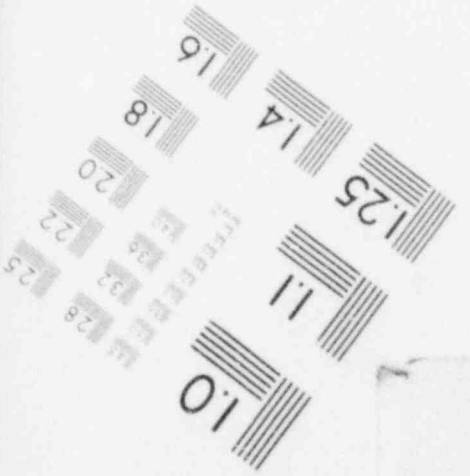

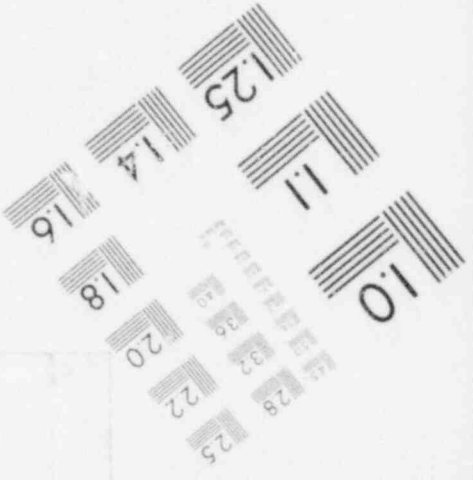

the sphere. The sample points were tested satisfactorily on May 2, 1974. This action eliminates RO's concern regarding the sampling of the East and West Pipeways for primary system leaks, originally brought to the licensee's attention on February 1971.[2/]

b. Safety Valves - Unit 3

All eight safety valves were replaced during the outage with valves set with steam at the new safety valve testing facility. The new testing facility is quite an improvement over the old nitrogen testing stand in that actual operating temperature and pressure can be simulated. The inspector witnessed the setting of two valves, both of which were found to be popping at a 12 to 16 psig lower than their previously set pressure using the nitrogen correlation.

The licensee did not keep records of the as found popping pressure of all the safety valves, therefore it is not possible to ascertain whether all valves popped prematurely and by what amount.

The newly set valves and their set pressures are as follows:

Serial #	Where Installed	Called Set Pressure	Actual Set Pressure (Two-Tests)
6282	B	1260 psig	1254 - 1246
6252	D	1260 psig	1252 - 1249
6299	F	1260 psig	1256 - 1246
6277	H	1260 psig	1260 - 1253
6304	C	1250 psig	1244 - 1237
6532	G	1250 psig	1244 - 1237
6294	A	1240 psig	1239 - 1236
6525	E	1240 psig	1242 - 1238

The average pressure of the two tests is in all cases within the $\pm$1% allowable deviation of the required set pressure.

The valve manufacturer has advised the licensee not to increase the set pressure above 1265 psig because the existing valve spring may reach yield stress and fail and also because the valve may not meet ASME Code requirement for a 5% blowdown. Both problems can be resolved, if needed, by replacing the springs for operation above 1265 psig.

c. Target Rock Safety - Relief Valve - Unit 3

One of the five electromatic relief valves (3A) was replaced with a Target Rock Safety - Relief. The set pressure for either solenoid

2/ RO Inspection Rpt No. 050-010/71-01.

Ibid.

or pressure actuation remained at 1125 psig. Four steam poping tests conducted by the vendor gave a set point range of 1123 - 1125 psig and a time delay of ~220/m-sec compared to 150 m-sec for solenoid operation. The valve was modified at the vendor shop to prevent disengagement of the second stage piston lock washer by axially drilling and pinning the nut to the shaft as specified by General Electric SIL #67.

4. Reactivity and Power Control

a. Unit 1 Temperature Coefficient

A temperature coefficient test was performed at a core average exposure of 13,164 MWd/t. The maximum temperature at which the coefficient was positive was approximately 300°F. The reactivity gain from ambient to 300°F was approximately 50 cents. Both valves are well within the Technical Specification limits of 550°F and 100 cents respectively. Core average exposure at time of refueling was 13,826 MWd/t.

b. CRD Scram Times - Unit 1

CRD testing was performed on October 15, 1973 in preparation for the outage. Scram times to buffer ranged from 1.800 second to 1.290 second. This is within allowable Technical Specification limit of 2.5 second maximum. Time in buffer ranged from 0.670 second to 0.140 second. Friction testing was also performed. Results ranged from 115 psi to 53 psi. In all cases CRD F-1 could not be tested because it was disabled. During the outage the following CRD's were removed and replaced with spare units: J-4, C-8, J-9, A-5 and K-7.

c. CRD Problems - Unit 1

(1) CRD F-1

As noted in previous inspection reports[3]/ CRD F-1 was stuck in the full in position until its blade and fuel were removed during the current refueling outage. Following reloading of the core the drive became stuck again, but the problem, which was caused by a cap screw lodged in the north wing of the guide tube, has been resolved. The event was reviewed during this inspection and found to be as stated by the licensee in his letter to DL dated May 10, 1974.

3/ RO Inspection Rpts No. 050-010/72-03, 73-02 and 73-06.

(2) CRD B-2

During the recovery of an orifice lodged in fuel cell 55-08, adjacent to CRD blade B-2 the "J" hook used in the operation was lost. (See Paragraph 5.b) Following the reloading CRD B-2 would not withdraw. The event was reviewed during this inspection and found to be as stated by the licensee in his letter to DL dated May 10, 1974.

(3) CRD J-3

Following the return of CRD B-2 to an operational status, CRD J-3 would not withdraw from its fully inserted position. The licensee again blamed crud in the mechanism as the cause. Following flushing operations the drive became operational again on May 13, 1974. The problem experienced with J-3 and B-2 appear to be similar in nature. As of the conclusion of this inspection all 80 drives were operational.

(4) Scram Outlet Valve Failure

On April 9, 1974 during scram testing of the CRD the scram outlet valve associated with accumulator No. 25 failed to open. The event was reviewed during this inspection and found to be as stated by the licensee in his letters to Directorate of Licensing dated April 19, 1974 and April 30, 1974.

d. Shutdown Margin - Unit 1

The licensee will perform shutdown margin checks during the startup phase of cycle IX. Calculations show that with all rods in, at BOC, the reactor is shutdown by $\sim$8.4%ΔK. The rods of higher worth are G-10 (peripheral) 3.9%ΔK, and F-9 (interior) 2.9%ΔK. The calculated shutdown margin with G-10 out is 4.3%ΔK/and with F-9 out is 5.3%ΔK. The valves are well within the limits set forth in the Technical Specifications.

e. CRD Blades Pull Test - Unit 1

In accordance with Technical Specification requirements the licensee conducted a pull test on all CRD blades, to ensure proper latching, as they were installed.

f. Shutdown Margin Checks - Unit 3

In accordance with the licensee's letter to Directorate of Licensing dated October 11, 1973 shutdown margin checks were performed on January 23, 1974, following shutdown of $>$72 hours duration. Ten

blades were checked and a shutdown margin in excess of 1%ΔK determined. Quadrant criticals similar to those obtained last August 1973 were attempted, but with the core near end of cycle life the reactor remained subcritical.

5. Core and Internals

a. Fuel Assembly UN-350 - Unit 1

An engineering evaluation of the acceptability of fuel assembly UN-350 was performed by the licensee. The assembly scratches were measured using an optical microscope and determined to have a maximum depth of 0.002 ± 0.001 inch which is within the acceptable range of $\leq$ 10% of clad thickness. The assembly has been loaded in the core.

b. Loss of "J" Hook in Reactor Vessel - Unit 1

During recovery operations of a orifice lodged in fuel cell 55-08 the licensee lost the stainless steel J-hook initially used. A review of the event revealed that:

(1) The J-hook used was 3/8" diameter approximately 13" long with an eye of approximately 1" diameter, fully closed but not welded together.

(2) It was tied to $\sim$1/8" diameter stainless steel cable secured with two cable clamps.

(3) The hook was lowered to recover the orifice (weight $<$ 10 pounds) but because the orifice was cocked it could not be raised with minimal force applied.

(4) The refueling crew wrapped the cable around a steel bar at the refueling bridge and with two men proceeded to apply maximum force on the hook, orifice, etc.

(5) The cable was recovered intact, clamps still in place, but hook missing. Therefore it has to be assumed the force applied had to be sufficient to open eye of hook.

(6) The force applied was calculated by Regulatory Operations to be in excess of 1,000 pounds.

(7) The licensee did not measure the force applied nor calculated it. It was found that the licensee does not use any kind of force measuring device when working with reactor internals.

(8) No procedure was used in the operation because the licensee considers such operations to be within the definition of "craft capability" although no such "capability" was shown by the operation.

(9) The licensee did perform calculations to show that the flow velocity below the lower grid plate cannot support the weight of the hook.

c. Jet Pump Inspection - Unit 3

As a result of the problems experienced with Quad Cities Unit 1, the licensee inspected 10 of the Unit 3 jet pumps to assure mechanical integrity. No discrepancies or abnormalities were noted.

d. Lower Tie Plates - Unit 3

The tie plates of 16 fuel assemblies suspected to contain surface hair line cracks were borescope inspected by the licensee and General Electric Company after brushing to remove possible crud film. The licensee reported good visibility, with the borescope kept within 2" of the lower tie plate. No evidence of cracking was noted.

e. Core II Fuel Performance - Unit 3

During in-core and out-of-core sipping of the 724 fuel elements the licensee identified 33 as leakers.

A review of the exposure history of all the elements in the core revealed that core II failure rate was as follows:

Exposure History, MWd/t	Total Elements	Leakers	% of Failure
<7000	120	0	0%
7000 - 7999	36	1	3%
8000 - 9999	522	27	5%
≥10,000	46	5	10%

The lead element at the end of cycle II had accumulated 10,307 MWd/t. Of the leakers, the lowest and hightest exposure history was 7,546 MWd/t and 10,164 MWd/t respectively.

In addition to the 33 leakers the licensee replaced an additional 11 elements for reactivity considerations. All new elements (44) used are of the new 8 x 8 design.

f. Second Reload - Unit 3

A review of the second reload for Unit 3 showed that 44 new fuel assemblies of the 8 x 8 type were loaded in accordance with the licensee's submittal to DL dated November 27, 1973. No new 7 x 7 assemblies were utilized.

g. Feedwater Sparger Inspection - Unit 3

A second inspection of the Unit 3 feedwater sparger was conducted April 14 to 18, 1974. Similar to last year's inspection[4]/ no signs of cracks were noted, and all components were reported to be in good condition. No video tapes were made of this inspection.

6. Electrical Systems - Unit 1

a. Station Batteries

The station batteries surveillance test records were reviewed as noted below and found to satisfy the requirements of the Technical Specifications.

(1) Weekly - January 1 to March 31, 1974.

(2) Quarterly - First quarter 1974.

(3) Every refueling outage - Eighth refueling outage.

The load discharge test, although satisfactory, was not performed in accordance with the temporary procedure in that data required by step No. 9 was not obtained. This is a violation of paragraph 6.2.A.7 of the Technical Specifications. Another deficiency noted was that the temporary procedure used for the test had not been dated and it is not possible to verify whether it was approved prior to or after the performance of the test.

A review of 9800-AN procedures revealed the licensee has not corrected his procedures to include the point in time, after the loss of the battery chargers, when a controlled plant shutdown would be initiated. This item has been outstanding since August 1971. In addition no effort has been made to generate the emergency procedures required in the event all d.c. power was lost (batteries and chargers). This item has been outstanding since April 1973. It is obvious that Unit 1 procedures do not yet comply with Regulatory Guide 1.33 although the licensee stated in a letter to the Directorate of Licensing dated March 1, 1973, that it would comply by March 1, 1974.

4/ RO Inspection Rpt No. 050-249/73-06.

b. Auxiliary Electrical Systems

The availability of auxiliary power to the unit prior to startup on June 20, 1973 (last startup prior to refueling) was reviewed and found to satisfy the requirements of the Technical Specifications.

7. Containment

a. Torus and Drywell Inspection - Unit 3

Sargent and Lundy Engineers and the licensee conducted an inspection of the drywell internal surface and of the torus surface above water line. The torus was not drained during this outage. The following findings and recommendations were made by the inspectors:

(1) No repairs are required inside the torus, vents and vent header.

(2) The drywell area above gallery at elevation 576 feet - 7 1/8 inches and 562 feet 0 inches require removal of all peeled vinyl coating and recoating.

(3) Steel surfaces on reactor head insulation and platform require removal of all peeled vinyl coating.

The licensee plans to scrape all peeling paint during this outage, but will not recoat the surfaces until the next refueling outage. According to the licensee Sargent and Lundy Engineers agree the planned actions are satisfactory.

b. Torus to Drywell Vacuum Breakers - Unit 2 and 3

Following RO findings of October 1973[5]/ regarding the setting of the torus to drywell vacuum breakers position indicating switches the licensee modified the position indicating system. The results were inspected and although greatly improved it is apparent that with the existing equipment the reliability of the system is questionable. The inspector tested valve 3-1601-33 B, C, D, E and F, and 2-1601-32 E and F. It was found that:

(1) One switch on valve 3-1601-33E did not come in contact with the actuating arm when the valve was fully closed.

(2) The counter weight arm on valves 3-1601-33 B and C could be moved ~5 to 10° without moving the valve shaft. This meant that the valve could move in excess of the limit without tripping the indicating lights.

5/ RO Inspection Rpts No. 050-237/73-05 and No. 050-249/73-06.

(3) While the modification package for both units indicated that all valves would indicate open if the arms moved 3/64" during the post modification testing the inspector noted that the movement was very dependeant on how valve was closed, slowly, or by its own from a position 20° open. The difference could change the gap by as much as 1/8".

c. Drywell to Torus Vacuum Breaker Leak Test

On May 27, 1974, the licensee performed a satisfactory leak test of the torus vacuum breakers by pressurizing the drywell to 1.28 psi above torus pressure. The results indicate the leak rate to be approximately one tenth of the allowable.

d. Local Leak Rate Testing - Unit 3

The licensee performed LLRT of primary containment testable penetrations during the outage prior to the performance of the LLRT. The results of the tests, in scf/hr. are as follows:

	Penetration	As Found	As Left
(1)	Electrical	42.120	42.120
(2)	MSIV's	7.339	2.487
(3)	Bellows Seals	0	0
(4)	Double Gasketed	Not Done	29.617
(5)	Feedwater	3225.000	0.835
(6)	Other Isolation Valves	55.075	28.009
(7)	Personnel Airlock	Not Done	6.426

These results compare to the limits in the Technical Specifications as follows

		As Found	As Left
(1)	Double-Gasketed Seals	Not available	0.50
(2)	Testable Penetration and Isolation Valves	19 times	0.45

The above results are obtained by using the most restrictive leakage path in any penetration. In other words, the figures used are the smallest measured leak rate, in the case the leak rate for both valves in known, or 1/2 the measured rate when only the total for the pentration is known. If one assumes that for each penetration one valve is going to fail, and, the largest known leakage path is then used, the "as left" condition for testable penetrations and isolation valves is approximately 80% of the Technical Specification limit. The total "as left" leakage rate measured during LLRT was $\sim$109.5 scf/hr. which is equivalent to 0.224 wt%/day of the containment air at 48 psig.

It should be pointed out that the "as found" condition of five (5) double gasketed penetrations were not obtained. The penetrations in particular are:

(1) equipment door
(2) CRD removal
(3) Drywell head
(4) two torus manways
(5) five torus to drywell vacuum breakers

As a result of exact value of the primary containment "as found" ILRT can not be determined. As it turned out, this was immaterial in this particular case since the leak rate through the feedwater check valves was so large, but it could be significant in the future. The failure to perform the LLRT's is considered to be a violation of paragraph 4.7.A.2.a (2) of the T/S and of 10 CFR Part 50, Appendix J, Section III.

The inspector witnessed the performance of two LLRT:

(1) LPCI Containment spray valves 1501 -27A and 28A
(2) "B" MSIV's

During both tests the inspector noted the licensee did not follow the General Procedure for Local Leak Rate Testing, 38-1600-S-0 step 8 or the specific procedure for the penetration (38-3000-S-I Step F in the case of the MSIV) in that it failed to vent and disconnect the line used to pressurize the penetration and instead relied on the tightness of the test equipment valves. This is considered to be a violation of paragraph 6.2.A (7) of the T/S.

During the performance of the "B" MSIV test on May 16, 1974, in the "X"-area two maintenance personnel were observed by the inspector violating Rad Protection Standards in that both men came into the X - area without the required anti-C protection clothing and half mask. This is considered to be a violation of paragraph 6.2.D of the T/S. According to Rad Protection personnel 1/2 masks were required in the area because the loose contamination of the floor exceeded 5000 c/m/ft^2.

e. Integrated Leak Rate Test - Unit 3

(1) Results

An ILRT was performed on Unit 3 primary containment during the period May 29-31, 1974 after completion of the local leak rate tests, and repairs where needed. The initial attempt failed as the leakage rate appeared to exceed the allowable operational

limit, LTO, of 1.2 weight %/day. Therefore after 11 hours, and an average leak rate of ~ 1.15 wt.%/day, the licensee proceeded to look for leaks.

The primary containment "as found" condition is therefore the ILRT results plus the difference between the "as found" and "as left" ILRT results = 1.15 + (6.88 - .22) = 8.25 wt.%/day or ~ 5 times the maximum allowable leakage rate of 1.6 wt.%/day.

Even if the feedwater check valve problem is disregarded the "as found" condition would not have met the maximum allowable operational leak rate of 1.2 wt.%/day.

On May 29, 1974, the licensee found the control air lines (36) which normally feed 25 psig air to the drywell cooler dampers to be a source of leakage. The leak was stopped by closing the manual valves (36) which isolate each line. The valves will remain closed during operation. They have also been closed for Unit 2. On May 30, 1974, another leak was isolated by cutting and plugging the communications conduit penetration which exits through the personnel airlock. A successful ILRT was then performed during the period of 1330 on May 30 to 1330 May 31, 1974. A successful verification test followed. Throughout the test the inspector obtained independent raw data. This data was later reduced at the office to verify the licensee's initial findings. It should be noted that during the test the licensee did not reduce the data in accordance with the method outlined in ANSI N 45.4-1972, as required by 10 CFR Part 50, Appendix J, and by their committment to Licensing.[6]/ The inspector's data was reduced in accordance with ANSI N 45.4-1972. It is expected that once the licensee recalculates his results in accordance with the above requirements better agreement between the inspector's and licensee's results will occur.

	La as Determined From	
	Inspector's Data	Licensee's Initial Results
For an 18 Hour Period	0.600 wt.%/day	
For First 13 Hour Period		0.570 wt.%/day
For Last 11 Hour Period		0.960 wt.%/day

This leakage rate is within the Technical Specification limit Lto of 1.2 wt.%/day. Between the 0930 and 1030 hours readings

6/ Letter, Able to Ziemann dated 9/26/7[illegible].

on May 31, 1974, all temperature measurements (8 sensors) located in the upper levels of the drywell showed an abrupt increase of approximately 10°F. Throughout the duration of the tests these measurements continued to slowly increase in temperature while the remaining 11 sensors in the drywell showed a slight cooling trend. No explanation for this apparent stratification exists. One theory is that the fans used by the licensee to maintain proper air circulation in the drywell may have moved. However this cannot be proven since the licensee failed to note the exact orientation of the fans upon reentry in the drywell due to a foulup in instructions.

The verification test performed on May 31, 1974, consisted of superimposing a known leak rate of 0.04 wt.%/hour. The results, after subtracting the controlled leak rate, were:

	Inspector's Data	Licensee's Initial Results
For 5 Hourly Periods	0.480 wt %/day	
For 7 Hourly Periods		0.960 wt.%/day

The results are within the acceptability criteria of 10 CFR 50, Appendix J which requires the verification test results to be within 25% of the ILRT results.

(2) Test Procedure

Test procedure 38-1600-S-XVI was used during the test. Some confusion existed at the beginning of the test and several temporary procedure changes had to be processed. In particular it is interesting to note that the procedure steps for determining the water vapor pressure in the containment were not compatible with the equipment being used and required last minute procedure changes which created some confusion.

The procedure failed to address the concerns of the following ANSI N45.4-1972 requirements:

(a) Paragraph 7.4 concerning temperature measurements. As a result, the drywell temperature data reduction method utilized throughout the test consisted in the arithmetic average of all temperature sensors. The inspector pointed this out to the licensee but no interest was shown in weight averaging until the temperature in the upper volume of the drywell showed a considerable change over a one hour period. During the exit interview the licensee agreed to recalculate the test results in conformance with ANSI requirements.

(b) Paragraph 7.8 - Recording of Data. A dated log of events and observations was so poorly maintained that when one temperature sensor failed early in the test (May 29) the word never was received by the other two shifts. As a result throughout the entire test (May 30-31) one shift was averaging 19 sensors to obtain an average drywell temperature while the other two shifts were averaging 20 sensors. In addition the 0800-1600 shift on May 30, which discovered and corrected the communications penetration leak in the personnel airlock, had only one entry in the log which stated the test start time had been moved to 0730 hours. The inspector was unable to obtain details of the fix and time of the fix until May 31, 1974 since the control room logs were also silent on the subject.

(c) Paragraph 7.9, Computation of Leakage Rate, requires that a statistically average hourly leakage rate be obtained by a linear least square fit of the hourly measured results. The licensee felt throughout the test this was not necessary. A plot of the hourly results was maintained by the inspector since no one else showed any interest in doing so. This deviation from ANSI requirements showed the licensee was unaware that it had been committed to do so by a letter to Licensing dated September 26, 1973.

All of the above deviations from ANSI N45.4-1972 are considered to be violations to 10 CFR Part 50, Appendix J.

(3) Equipment Used

The twenty drywell and four torus thermocouples were checked against a test thermometer which had been calibrated on May 3, 1974 by the Operations Analysis Department. An average drywell temperature correction of -1.1°F and a torus temperature correction of +1.0°F were obtained. The temperature read out had to be changed prior to the test because the range of the originally planned recorder read out was less than the temperatures experienced inside the drywell (>100°F). The readability of the used indicator was 0.5°F. No records of its accuracy were available.

Absolute sphere pressure was monitored by a 0-330 Cm Trimount absolute manometer which had been calibrated on May 8, 1974, against a Wallace and Tiernan barometer. The latter had been calibrated against the barometer in use at Midway Airport. The manometer with divisions of 0.1 Cm had a vernial dial which, when operable, gave readability to 0.01 Cm. Throughout the test the vernial dial was only partially operable and the probable readability was closer to 0.05 Cm.

Relative humidity in the torus was monitored by two Foxboro dewcell instruments which read in ohms. No calibration records were available. Relative humidity in the drywell was monitored by four sensors. Two were the permanently mounted sensors which read on the environmental rack and which read directly in percent relative humidity. The other two were temporarily installed General Eastern which read in volts and had been previously used in the Unit 1 sphere ILRT last January. No calibration records existed for the permanent instruments and the only records available for the temporary instruments were for the calibration performed prior to the Unit 1 sphere test.

The lack of properly calibrated sensors is a violation of 10 CFR Part 50, Appendix B, Criterion XII and XVII.

8. ECCS - Unit 1

a. Core Spray Pump "B" Breaker Trip

On December 27, 1973 the "B" core spray pump tripped during a logic test. The event was reviewed during this inspection and found to be in accordance with the licensee's letter to the Directorate of Licensing dated January 4, 1974, with the following exception:

> The Westinghouse breaker in question has a slow trip of 300 amps and not 1500 amps as stated in the letter. The inspector was not able to verify the stated fast trip of 3000 amps.

The corrective actions listed in the letter were performed by January 16, 1974. The quality control group has no documentation that a like-for-like replacement was installed. This breakdown in proper documentation occurred because the maintenance group failed to keep the quality control group informed of what they were doing. This continues to be a major deficiency in the implementation of Quality Assurance at the site. This deficiency in documenting the qualification of the replacement part or equipment is considered to be a violation of 10 CFR Part 50, Appendix B, Criterion XVII.

b. "C" Core Spray Pump Line Leak

On January 10, 1974, the licensee experienced a leak in the "C" Core Spray pump minimum flow line. The event was reviewed during the inspection and found to be as stated by the licensee in his letter to Directorate of Licensing dated February 8, 1974. Thickness measurements were made on lines "A" and "B" and found to nominally meet the thickness requirements for 2" sch. 80 pipe. The modification to install air operated valves in the lines was sent to

M&S Engineering on January 18, 1974. It appears to be stalled there, as no action has been taken since. The licensee stated that an evaluation of ECCS system is in progress to determine whether to change from river water to condensate water, and this major study may have caused the delay in the modification referred to in their letter to Directorate of Licensing.

c. Core Spray Valve Interlocks

The licensee has performed a modification (M-12-1-73-15) which interlocks valves CS-16, 32 and 33 so that CS-16 cannot be opened unless CS-32 and 33 are closed. In addition CS-32 and 33 cannot be opened unless CS-16 is first closed. This prevents the possibility of discharging contaminated water to the storm sewer during the post incident operation mode. The modification completed on February 8, 1974 was performed in accordance with Quality Assurance requirements.

Modification M12-1-74-11 calling for a second valve in series with CS-16 is still under study at M&S Engineering.

9. Other Engineered Safety Features

a. Standby Liquid Control System - Unit 1

(1) Modifications

A test line between valves AO 300 and 301 was installed in accordance with modification M12-1-73-79 to facilitate testing the operability of the system. Following installation the licensee performed the test to prove that the line from the boron tank to the reactor is open.

A review of the modification package revealed that it had been signed authorizing operation of the modified equipment on January 31, 1974. The records showed that the site personnel failed to recognize the requirements of USAS B31.1 in that paragraph 137.1 requires either a hydrostatic leak test, an initial service leak test, a vacuum test or 100% radiography. Failure to comply with such requirement is considered a violation of Quality Procedure Q.P. No. 3-51, Revision 2, Sept C.17 in that the Operating Engineer authorized operation of the equipment, and the deletion of the 1.5 times design pressure hydro test, without M&S Engineering Department concurrence.

(2) Surveillance Testing

In accordance with existing Technical Specification boron concentration tests were performed on February 3, 1974 and May 16, 1974. The results showed that the minimum requirement of 400 pounds of boron in the system has been satisfied.

(3) Failure of B Pump

On March 1, 1974, the licensee experienced the loss of the "B" standby liquid control pump due to a frozen crankshaft. The event was reviewed during this inspection and found to be as stated by the licensee in his letter to the Directorate of Licensing dated March 28, 1974. The heat tracing modification mentioned in the letter has been approved by the M&S Engineering Department. Installation will proceed as soon as the material is received.

b. Shock Suppressors - Unit 3

All 31 shock suppressors located inside Unit 3 drywell have been fitted with ethylene-propylene seals and O-rings. The work was completed on May 14, 1974, and a review of the records showed compliance with Quality Assurance requirements.

10. Emergency Power - Unit 1

a. Surveillance Requirements

The Unit 1 diesel generator and Units 1 and 2 diesel driven fire pumps surveillance test records were reviewed for the period of January 1974 to April 1974 and found to satisfy the requirements of the Technical Specifications.

The diesel generator auto start and load acceptance test required to be performed every refueling outage had not been performed as of the time of this review.

b. Diesel Generator Breaker to Bus 16

On November 29, 1973, the licensee experienced a failure to close of diesel generator breaker to bus 16. Breaker to bus 15 operated satisfactorily. Bus 16 could be energized, if needed, from bus 15 through 5A and 6A. On October 30, 1973, both breakers had been tested satisfactorily. The fuel was out of the core at the time of the failure. On December 6, 1973, the breaker was inspected and overhaule No faults were found. On February 28, 1974, the breaker was cycled 10 times successfully. The licensee plans to replace the breaker if it fails again.

c. Diesel Generator Non Essential Load Lockout

In a modification completed on April 8, 1974, the licensee has locked out the following loads during auto start of the D/G:

(1) Screen refuse pump.
(2) Jockey fire pump.
(3) Demineralized water jockey supply pump.

In addition the following loads are disconnected but can be restarted at the discretion of the operating staff:

(1) A and B reactor enclosure cooling water pumps.

(2) A and B bearing lube water pumps.

The modification was reviewed and found to satisfy the Quality Assurance requirements.

11. Refueling - Unit 3

The following refueling surveillance test records were reviewed and found to satisfy the requirements of the Technical Specifications.

a. Refueling interlocks weekly test - March 17 to April 26, 1974.

b. SRM functional daily test - March 17 to May 3, 1974.

c. Fuel pool level recorded daily - March 4 to May 5, 1974.

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RO Inspection Report No. 050-010/74-06

A. RO Inspection Report No. 050-249/74-05

Transmittal Date : August 6, 1974

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