U. S. ATOMIC ENERGY COMMISSION
DIRECTORATE OF REGULATORY OPERATIONS

REGION III

Report of Operations Inspection

RO Inspection Report No. 050-263/74-02

Licensee: Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

Monticello Nuclear Generating Plant
Monticello, Minnesota

License No. DPR-22
Category: C

Type of Licensee: BWR (GE) 545 Mwe

Type of Inspection: Routine, Unannounced

Dates of Inspection: March 5 - 8, 1974

Dates of Previous Inspection: February 27 - March 1, 1974 (Radiation Protection)

Principal Inspector: P. H. Johnson — 4/8/74 (Date)

Accompanying Inspector: H. C. Dance

Other Accompanying Personnel: None

Reviewed By: H. C. Dance, Senior Reactor Inspector BWR Operations — 4/9/74 (Date)

SUMMARY OF FINDINGS

Enforcement Action

The following violations are considered to be of Category II severity:

A. Technical Specification 4.6.C.1 requires that a reactor coolant sample be taken to determine (1) a gross beta activity at least every 96 hours, and (2) an isotopic analysis at least once per month.

Contrary to the above (Paragraph 6.a):

1. The reactor coolant was not analyzed for gross beta activity between the period February 14 and 21, 1974.

2. An isotopic analysis of the reactor coolant was not performed during November 1973.

B. Technical Specification 4.6.C.2 requires that during steaming rates below 100,000 pounds per hour, a sample of reactor coolant be analyzed every four hours for conductivity and chloride content.

Contrary to the above, such analyses were not performed during the interval of February 16 - 18, 1974, while steaming at less than 100,000 pounds per hour. (Paragraph 6.a)

Licensee Action on Previously Identified Enforcement Matters

The licensee has completed corrective actions related to items 5.a, 6 and 9.b as identified in the RO:HQ enforcement letter following the May 1972 management audit. (Paragraphs 4 and 5)

Unusual Occurrences

A. An RCIC steam line high area temperature switch was found on January 29, 1974, to have drifted outside its allowed limiting setpoint. (Paragraph 12)

B. The "A" RHR torus cooling injection valve operator motor failed on February 3, 1973, due to overheated motor windings. (Paragraph 11.f)

C. Two main steam isolation valves failed to close during a routine surveillance test on February 16, 1974. (Paragraph 9)

Other Significant Findings: None.

Management Interview

The inspectors conducted a management interview with Messrs. Neils (NSP General Superintendent, Nuclear Power Plant Operation), Larson (Plant Manager), and supervisory member of the plant staff at the conclusion of the inspection. The following matters were discussed:

A. The unusual occurrences reviewed during the inspection and the licensee's related plans were briefly discussed. (Paragraphs 9, 11.f and 12)

B. The inspector discussed his review of activities related to the off-gas system, noting that he had no comments related to the conduct of the preoperational testing program. He stated that further review would be given to planned retreatment of the A recombiner vessel and modification of recombiner heater control circuitry. (Paragraph 14)

C. The inspector stated that based upon review of the licensee's related corrective actions, violations from the May 1972 management audit related to Operating Manual review and Volume F Memos were considered to have been corrected, but that a followup examination of these areas would be conducted in late 1974. The inspector also stated that in view of the licensee's retraining program that had been submitted to Licensing, the related violation from the same audit was also considered to have been corrected. (Paragraphs 4 and 5)

D. The licensee was reminded to ensure that APRM flow-biased scram setpoints, after correction for APRM gain, remain within Technical Specification limits. (Paragraph 7.c)

E. The violations involving the omission of a reactor water isotopic analysis during November, and conductivity and chloride requirements during low steaming rates in February were identified. The licensee acknowledged the findings. Subsequently, the inspector informed the licensee of the absence of the gross beta activity analysis during the same February period. (Paragraph 6.a)

F. The inspector stated that the Technical Specifications listed the HPCI discharge pressure range as 150-1150 psig although Technical Specifications Change Request No. 3 had requested a change to 1120 psig. This request has since been cancelled. The licensee stated that a request for Technical Specification change would be resubmitted. The licensee stated that following test line modifications during the scheduled outage, the HPCI discharge pressure would be demonstrable over its full range. (Paragraph 11.d)

REPORT DETAILS

1. Persons Contacted

   C. Larson, Plant Manager
   M. Clarity, Superintendent, Plant Engineering and Radiation Protection
   W. Anderson, Superintendent, Operation and Maintenance
   L. Eliason, Radiation Protection Engineer
   G. Jacobson, Plant Engineer, Technical
   D. Antony, Plant Engineer, Operations
   S. Pearson, Shift Supervisor
   B. Day, Engineer
   F. Fey, Assistant Radiation Protection Engineer
   M. Hammer, Engineer
   J. Heneage, Engineer
   W. Hill, Engineer
   R. Jacobson, Plant Chemist
   B. Jenness, Engineer
   D. Nevinski, Engineer, Nuclear
   L. Nolan, Engineer
   J. Pasch, Engineer
   R. Perry, Engineer
   W. Shamla, Engineer, Instruments

2. General

   The Monticello plant was operating at a reduced power level of 76% at the time of the inspection to maintain stack release rate below an administrative limit of 100,000 uCi/sec. The plant was scheduled to shut down on March 14, 1974, for a refueling outage of approximately 11 weeks' duration.

3. Log and Records Review

   The following records were examined during the inspection without comment:

   a. Reactor and Control Room Log - February 16 - 20, 1974.

   b. Operations Committee Minutes - October 10, 1973 - January 16, and February 1, 14 and 15, 1974.

   c. Safety Audit Committee Minutes - November 9, 1973 and January 9 - 10, 1974.

   d. Weekly battery readings for No. 13 250 volt battery, October 30, 1973.

4. Retraining Program

Item 6 of the enforcement letter[1] following the May 1972 management audit identified certain aspects of the retraining program which did not comply with Technical Specifications requirements. The inspector noted during the inspection that this violation had been corrected by the formal retraining program which was submitted[2] by the licensee to the Directorate of Licensing in response to Appendix A to 10 CFR 55.

5. Operating Procedures/Volume F Memos

Item 9, Part b, of the enforcement letter[3] following the May 1972 management audit noted that semiannual reviews of the operations manual had not been completed as required. The licensee's response[4] to the enforcement letter stated that a new review schedule had been established which would become effective following the first rewrite of each manual section. Examination of manual review records by the inspector showed that 78 of the 100 manual sections had been revised and reviewed by the Operations Committee, with most of the remainder in progress. A representative stated that reviews of several of the sections of Volume A, General Administration, were being held in abeyance pending issue of the Administrative Controls Manual, which will supplant significant portions of the present Volume A. A spot check of the status list maintained for manual revisions against Operations Committee minutes revealed no discrepancies. Periodic reviews subsequent to the initial review were noted to be proceeding on schedule. A representative stated that an additional change to the review schedule had been approved by the Operations Committee, to the effect that the routine periodic review of radiation safety procedures, Volume E, had been changed from annual to biennial, except for E.2 (Emergency Plan), which would continue to be reviewed annually.

Item 5, Part a, of the enforcement letter[5] cited noncompliance related to temporary changes to operating procedures. Review of the master copy of Volume F Memos kept in the control room showed 84 to be in effect as compared to 206 in November 1972. Some of those remaining in effect were to be deleted by pending revisions to the Operations Manual. A February 1974 revision to Section A.6, Plant Operating Practices, of the Operations Manual was noted to have provided more detailed guidelines for the

1/ Letter, RO:HQ to NSP, dated 10/19/72.
2/ Letter, NSP to DOL, dated 12/17/73.
3/ Letter, RO:HQ to NSP, dated 10/19/72.
4/ Letter, NSP to RO:HQ, dated 11/10/72.
5/ Letter, RO:HQ to NSP, dated 10/19/72.

review, approval, and issuance of Volume F, Temporary Memos. Three categories of memos--description, orders and procedures--are defined, the latter two of which require Operations Committee approval within thirty days. Cancelled Volume F memos were noted to have been removed from the control room copy, although several inconsistencies in the index and the manual chapter cross-reference list were noted.

The inspector stated that based upon improvements shown by the licensee in the review of operating procedures and Volume F memos and in view of the significant reduction in the number of Volume F memos in effect, the related violations were considered to be resolved, although a followup review of the two areas was planned for late 1974.

6. Reactor Coolant System

a. Coolant Chemistry

Review of reactor coolant analyses between December 31, 1973 and February 21, 1974 (except as noted) determined the following with respect to Technical Specification 4.5.C and 4.6.C:

(1) A reactor coolant sample was not taken at least every 96 hours as required by Technical Specification 4.6.C.1(a) between February 14-21, 1974, and analyzed for gross beta activity.

(2) A reactor coolant sample was not taken every four hours as required by Technical Specification 4.6.C.2 and analyzed for conductivity and chloride content between February 16-18, 1974, while steaming at less than 100,000 pounds per hour.

(3) An isotopic analysis of the reactor coolant system was not performed during November 1973 as required by Technical Specification 4.6.C.1(b).

The surveillance data sheet on November 20, 1973 indicated an isotopic analysis (gamma scan) had been performed. Discussions with plant personnel established that this data sheet is initiated when the gamma scan is initiated. Due to a laboratory mix-up the sample was not counted long enough and the work was never completed. The licensee's system of monitoring required surveillance testing had not detected the above omissions.

Review of analyses from October 1973 - February 1974, indicated typical values were as follows:

Total Iodine	:	3.2 (Nov) - 1.65 (Feb) uCi/ml
Chlorine	:	10 ppb
Conductivity	:	0.14 - 0.7 umho/cm
Gross Beta	:	1.3 - 2.2 uCi/ml

b. Coolant Leakage

Reactor coolant leakage rate was confirmed to be set up to complete on daily basis. On February 18, 1974, the leakage calculation was confirmed to be within the limits of Technical Specifications 3.6.D and 4.6.D and as also indicated on summary data sheets for the period 12/1/73 to 1/31/74.

Review of the recently installed reactor vessel leak detection system showed it to be operating essentially as described in a licensee letter.[6] Continuous recording and indication of floor drain and equipment drain sump levels were noted to be available to the operator, plus a computer point which permit readout of sump level and rate of change (in GPM) at any time, based upon computer inputs at 15-second intervals. A licensee representative stated that some downward shift in the indicating range of the floor drain sump level indicator had been observed, with the effect that the indicator goes off scale low following pump-down of the sump. Refinement of the indication was planned for the forthcoming refueling outage. The representative stated that system performance had otherwise been good, and that no difficulties had been experienced in the operation of the sump pumps and their controlling float switches. The floor drain leak rate indicated by the process computer, as confirmed by observation of the floor drain sump pump run frequency, was noted to be 0.09 gpm.

c. Other Surveillance

The recirculation system cross-tie interlock check required by Technical Specification 4.5.I.1 was confirmed to have been satisfactorily completed monthly from November 1973 through January 1974.

Reactor safety and relief valves were confirmed from a review of surveillance tests to have been tested and inspected as required by Technical Specification 4.6.E. All safety valves were set at 1240 psig and relief valves at ≤ 1069 psig during the October 1973 outage as identified in RO Inspection Report No. 050-263/73-11. Relief valve bellows leakage tests were confirmed to have been performed each three months between July 1973 and January 1974.

7. Reactivity and Power Control

a. Control Rod Drives

Review of CRD scram times during the current cycle established that 49 CRD's remain to be tested to meet Technical Specification 4.3.C. Completion of this testing was confirmed to be scheduled immediately

6/ Letter, NSP to DOL, dated 12/28/72.

after shutdown for the refueling outage scheduled March 14, 1974. All data reviewed were taken from the multipoint recorder which connects 28 CRD's. All scram times were within times contained in Technical Specification 3.3.C. On November 6 the maximum 90% scram insertion time was 2.84 seconds.

Correlation between the multipoint recorder trace and individual rod insertion traces from the brush recorder was established. Included in this review was the testing performed as a result of GE questioning on September 28, 1973, the fast CRD insertion times from full out to 5% insertion. Subsequent testing by NSP determined that 85 msec should be added to the scram times. This interval was the demonstrated time delay that it takes pen No. 30 of the multipoint recorder to buildup to a printing threshold of 1.68 VDC. Data since October 1973 were stated to be corrected by adding 85 msec. Review of the November 6, 1973, was inconclusive to the inspector since the starting point was not clear. Even considering the above correction the CRD's meet the required specifications.

CRD stall flow testing during January and February 1974 indicated two drives with greater than 5 gpm. Approximately 25 CRD's were scheduled to be replaced during the outage on the basis of past tests and routine change out. Two drives scheduled for replacement are 18-31 and 22-35 which have been in service since startup and are the only two modified drives with the inner screen mounted on the stop piston. No plans exist for modifying other CRD's since scram times continue to be satisfactory.

Weekly control rod exercise tests required by Technical Specification 4.3.A.2 were reviewed for the period December 29, 1973 - February 23, 1974, and confirmed to have been completed.

Status of control room accumulator level and pressure alarms was confirmed to be included in the Daily Surveillance Log. Compl^tion was verified . - December 18, 1974.

b. Intermediate Range Monitors (IRM's)

The functional test of the SRM rod block and the IRM scram and rod block were confirmed to have been performed as required by Technical Specification Table 4.1.1. on February 16, 1974, prior to the reduction to low power operation. The IRM requirements were confirmed for the outage bounded by the November 13 and November 16, 1973 test. Discussions with instrument personnel confirmed that the deviation values in parenthesis on IRM test No. 0013/0043 were being used as permissible drifts and not as "as left" settings. Instrument Department as left records indicated the IRM values of Technical Specification Table 3.1.1 were satisfactory from July 30, 1973 to February 16, 1974.

The IRM heat balance calibration was confirmed to have been completed for the shutdown and subsequent startups beginning November 14, 1973 and February 16, 1974. The calibration appears difficult with questionable accuracy at the low power with changing conditions. Settings were considered conservative.

Date	Heat Balance %	IRM	APRM
February 18, 1974	0.88	3	3
November 18, 1973	1.87	2.3-7	--

c. Average Power Range Monitors (APRM's)

APRM heat balance calibrations performed were reviewed for the interval January 2 - February 25, 1974 and found satisfactory. In gerneral APRM output signals were being left 0.5 - 1.5% higher than calculated thermal power. All APRM channels were noted to be conservatively set on March 5, 1974. Computer heat balances compared within 0.5% to manual heat balances for the review period February 1 - 15 and March 1 - 4, 1974. The calculations are confirmed three times per week.

APRM weekly functional scram tests for the period November 1973 - January 1974 required by Technical Specification Table 4.1.1 were reviewed and found satisfactory.

Calibration data for APRM flow-biased scram and rod block were reviewed against Technical Specifications requirements, and were noted to comply when the conservative APRM gain settings used by the licensee were taken into account. The APRM flow-biased scram setpoint corresponding to 50% recirculation driving flow was noted to have been left at 87.6-87.9% for all channels on January 24, 1974, but APRM indicated power was noted to be generally 0.5 to 1.5% conservative during this period. The licensee was reminded to ensure that the trip points, after adjustment for APRM gain, remain within Technical Specifications limits.

d. Core Checks

Reactivity anomaly checks were confirmed to have been performed monthly from June 1973 through March 1974 by comparing rod insertion values to GE provided curves. Core reactivity has continued to decrease as predicted since the cycle 2 startup.

The peak heat flux determination was confirmed to be setup routinely on a daily basis as required by Technical Specification 4.1.B. On February 1, 1974, at 82% power the peak heat flux was calculated to be 289,000 BTU/hr-ft^2 with a peaking factor of 2.7. The latter was determined by using conservative type curves provided operating personnel. Computer calculation indicated the peaking factor to be 2.2.

e. Other Surveillance

The Main Steam Line Isolation Valve Closure Scram Test Procedure (Test No. 0008) was reviewed to assure that each of the contacts and relays are tested. The test required by Technical Specification 4.1.1 was found satisfactory.

Reactor High Pressure scram settings were found to have been set properly between November 1973 and February 1974 as required by Technical Specification 3.1.A and 4.1.A. The bases for the procedural setpoint value was confirmed by tracing system elevations.

f. RO Bulletin No. 73-6

The above bulletin dated November 27, 1973 requested licensees to describe their administrative control system of coordinating core movements to prevent an inadvertent criticality. The licensee's response dated January 10, 1974, was determined to be satisfactory based on previous reviews of procedural and administrative controls. As stated in the response, the licensee verbally indicated that improvements in coordinating core manouvers were being reviewed.

8. Refueling Preparations

A facility representative stated during discussions with the inspector that the new fuel to be inserted into the core during the refueling outage had been inspected and placed in the new fuel storage vault. He explained to the inspector new procedures which will use the assistance of a computer to prepare the sequence used in the moving of fuel and other core components during the outage.

The inspector reviewed reports of inspections of the reactor building crane and refueling platform conducted in October 1972, by a crane inspector from an outside firm. The reports indicated no unsatisfactory conditions. A facility representative stated that a similar inspection by the same firm had been conducted during the last week of February 1974, and that a report was expected in the near future. The inspector examined procedures for semiannual inspection of the reactor building crane which were being sent to the Operations Committee for review prior to issue. Requirements for magnetic particle testing of the main hoist hook and dye penetrant test of the auxiliary hoist hook were to be added. The inspector questioned the licensee's fulfillment of the commitment for semiannual crane inspection expressed in Section 10.2 of the FSAR. The licensee representative referred to the inspections by the outside firm prior to the lifting of heavy loads during the previous and the forthcoming refueling outages and stated that issue of the new procedures would provide for semi-annual inspections thereafter.

9. Failure of Two Main Steam Isolation Valves (MSIV's) to Close

A licensee report[7/] discussed the failure of the outboard MSIV's in the B and C main steam lines to close during a routine surveillance test on February 16, 1974. The report also discussed corrective actions taken, including repairs performed on the air solenoid valves of all 8 MSIV's. Discussions with licensee representatives and review of a schematic diagram showed that (1) a differential pressure (equal to instrument air pressure) is normally applied across the viton seat of the AC solenoid, (2) air pressure is normally applied to both sides of the DC solenoid valve, such that no differential pressure exists, (3) seat deformation on the DC solenoid was not significant, (4) based on recommendations from the vendor, spring-loaded seats were not installed in the DC solenoid valves, and (5) the metal chips discussed in the licensee's report were not considered to be a significant factor in the malfunctions observed. Surveillance test records showed that all MSIV's had closed in the required 3-5 seconds during tests following the repairs. The inspector asked whether spring cushioned upper seats might eventually be necessary on the DC solenoids, but otherwise had no comments on the corrective actions taken by the licensee. A licensee representative stated that overhaul of all solenoid valves under the supervision of a vendor representative was planned during the forthcoming refueling outage, and that further evaluation of solenoid valve performance would be made at that time.

10. Vane Type Flow Switches

A licensee representative stated during the inspection that the vane type flow switch installed in the standby liquid control system was to be removed during the forthcoming refueling outage, in that it is not required for proper system operation. He stated that the stubs of the original flow switch paddles were still operating satisfactorily in the residual heat removal system, although further improvements were being planned for this system and for the reactor water cleanup system. These would likely utilize annubar-type (pitotstatic) indicators, and would probably not be installed during the 1974 refueling outage.

7/ Letter, NSP to DOL, dated 2/25/74.

11. Emergency Core Cooling System

Review established that test procedures, most of which have been rewritten in an improved format, are provided for each of the required surveillance tests. Bases for many of the procedural setpoints were not readily available at the station or in the procedure; for instance, the core spray pump discharge pressure equivalent to a reactor discharge pressure specified in the Technical Specifications. The latter was determined to be satisfactory from discussion with plant personnel and review of test results prior to initial reactor startup. Other systems values appeared reasonable but were not rechecked.

The RHR sub-system and HPCI were confirmed to be properly set-up on the control room panels during the inspection. Equipment in the RHR sub-system pump rooms was noted to be satisfactorily lined up.

a. Low Pressure Coolant Injection (LPCI)

Testing requirements contained in Technical Specification 4.5.B.1 were confirmed to have been performed for the intervals noted: quarterly flow rate, November 1973 - February 1974; pump operability, June 1973 - February 1974; and MOV operability, October 1973 - February 1974.

CV-1995, No. 12 pump minimum flow valve, closed automatically on low flow during testing on January 2, 1974. From the work request form and discussion with plant personnel it was established that the low flow indicating switch was loose, resulting in a calibration shift. The valve remained operable in the manual mode from the control room. Flow testing was satisfactorily performed following maintenance.

b. Core Spray Systems

Testing requirements in Technical Specification 4.5.A.1 were confirmed to have been successfully completed for periods shown: quarterly flow rate: January - February 1974 (three tests); monthly pump operability: October 1973 - February 1974; monthly MOV operability: July 1973 - February 1974; monthly header ΔP and calibration: October 1973 - January 1974 (four tests); and daily header ΔP check: October 31, 1973 and February 1, 1974.

c. Residual Heat Removal Service Water System

Quarterly flow rate test requirements contained in Technical Specification 4.5.C were found to have been satisfactorily performed for the June 1973 - February 1974 interval reviewed. Tests were noted to have been performed on the "A" system following maintenance (September 1973) and for a HPCI inspection(June 1973).

d. High Pressure Coolant Injection System

Surveillance tests designated in Technical Specification 4.5.D.1 were found to have been satisfactorily performed for the intervals reviewed as shown: monthly pump operability: September 1973 - February 2, 1974; monthly MOV operability: January - February, 1974; and quarterly flow rate: July 1973 - January 1974.

Difficulty has been experienced in simulating the required reactor pressure range during tests (typically values of 150 to 1120 are obtained) due to the oversized throttling valve (MO-2011) in the test line. The inspector reviewed a modification scheduled for the March 1974 outage to install a 6" self drag type valve to correct the difficulty. The pressure range of 150 - 1150 psig called for in the Technical Specifications was identified in Change Request No. 3 (dated August 20, 1971) as in error and should read 150 - 1120 psig according to the licensee. The licensee agreed to re-initiate action to revise the pressure range in the Technical Specifications. Following the above modification, the full designed flow and pressure range is expected to be demonstrated during each test.

The HPCI system's primary flow and pressure instrumentation, including system trips was noted to have been calibrated and checked on January 28, 1974, in accordance with test procedure No. 7130.

e. ECCS Instrumentation

Calibration of ECCS instrumentation listed in Technical Specification Table 4.2.1 was confirmed to have been conducted as required for the period November 1973 through February 1974. Once per cycle tests were conducted in May 1973.

Micellaneous core spray flow and pressure instruments were confirmed to have been routinely checked each six months since January 1973.

f. Torus Cooling Injection Valve

A licensee report[8/] described the failure of the "A" Torus Cooling Injection Valve, MO-2008, on February 3, 1974. A previous licensee report[9/] discussed a similar failure of the same valve, although the inspector concluded the causes of the two failures to be unrelated. (During preparation of the current inspection report, it was noted that the earlier licensee report had been incorrectly referenced in a

8/ Letter, NSP to DOL, dated 2/12/74.
9/ Letter, NSP to DOL, dated 11/7/73.

previous inspection report[10]). Review of the occurrence and a discussion with licensee representatives showed the events to have been as described in the licensee's report. The abnormal occurrence file contained a listing of ECCS motor operated valves which included 12 valves similar to MO-2008. Examination of one of these valves by the inspector during a plant tour showed the stem clamp to have been tightened and staked as indicated in the licensee's report. The licensee representative stated that a review of system diagrams indicated that no similar valves were located in the drywell, but that a review inside the drywell would be made during the refueling outage.

12. RCIC Steam Line Temperature Switch

A licensee report[11] discussed a condition wherein a steam line high area temperature switch associated with the reactor core isolation cooling (RCIC) system tripped at a temperature greater than that allowed by Technical Specifications. A similar occurrence (related to the high pressure coolant injection system) reported by the licensee was reviewed during a previous inspection.[12] It was noted during the earlier inspection that discussions between the licensee and the manufacturer were ongoing with relation to the drift experienced with temperature switches of this type. The licensee had also indicated in discussions with the inspector that more frequent calibration would not likely improve the performance of the switches, since they must be removed for calibration and the increased handling would likely offset the gains of increased calibration frequency. A licensee representative stated during the current inspection that consideration was being given to new temperature monitors, probably using thermocouples, which could be calibrated in place and which gave promise of more reliable operation. The representative stated that the switch which most recently malfunctioned had been removed from service and that the practice of setting the temperature switches at 10-15°F below their Technical Specifications limit would continue pending evaluation of an alternate installation. The inspector verified by review of calibration records that the as-found and as-left setpoints of all other temperature swithces associated with the RCIC system had been within Technical Specifications limits during calibrations performed on October 30, 1973 and January 29, 1974.

13. Hydaulic Shock Suppressors and Restraints

A previous inspection report[13] discussed actions taken by the licensee in response to Regulatory Operations Bulletin Nos. 73-3 and 73-4 and a

10/ RO Inspection Rpt No. 050-263/73-12.
11/ Letter, NSP to DOL, dated 1/30/74.
12/ RO Inspection Rpt No. 050-263/73-11.
13/ RO Inspection Rpt No. 050-263/73-12.

related letter from the Directorate of Licensing. A subsequent letter[14/] from the licensee described a reinspection of all suppressors within the drywell on February 17, 1974. An attached table also described repairs performed to suppressors located outside primary containment. The inspector examined surveillance documentation of February 17, in-containment suppressor inspection and of an inspection conducted on February 13 of suppressor units located outside of the drywell. No exceptions to the conditions reported in the licensee's letter were noted by the inspector. A licensee representative stated that suppressor units outside primary containment would be inspected once more prior to the refueling outage, and that a followup report was intended. He stated that all suppressor units had been reworked using internal soft parts made of ethylene propylene and new oil-fill fittings having Buna-N seats.

14. Off-Gas System

A review of off-gas system testing during the inspection showed that all preoperational tests had been completed. Operations Committee Minutes reviewed by the inspector indicated that all but four of the preoperational tests had been reviewed by the Operations Committee. A licensee representative stated that three of these four had been reviewed in recent meetings for which minutes were not yet issued, and that a final test report for the completed T3-1C ventilation system tie-in was soon to be received from the test engineer. The inspector reviewed several of the completed tests for general content and test results. The contents of each test package were noted to be as described in a previous inspection report[15/] The inspector made no comment on the portions of the system test program conducted to date.

Licensee representatives stated that plans called for tie-in of the off-gas system to the plant during the refueling outage, following corrective actions related to the recombiner and its heater control circuitry. Operational testing of the off-gas system would then be accomplished following plant startup after the outage. A corporate representative stated during a telephone discussion that metallurgical tests of the recombiner vessel material had been completed, leading to a conclusion that the vessel could be annealed in place to provide satisfactory performance. He stated that some reduction in yield strength of the lower portion of the vessel had occurred, but that the overthickness in the initial design would still provide adequate strength. The vessel was also to be hydrostatically tested to its initial test pressure following the in-place annealing process. The representative also stated that the heater control circuits had been redesigned and would be modified accordingly

14/ Letter, NSP to DOL, dated 2/15/74.
15/ RO Inspection Rpt No. 050-263/73-08.

during the outage. The inspector deferred comment on plans for the recombiner vessel and its heater controls pending review of the licensee's evaluation of the proposed corrective actions.

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A. RO Inspection Report No. 050-263/74-02

Transmittal Date : April 9, 1974

Distribution:
RO Chief, FS&EB
RO:HQ (5)
DR Central Files
Regulatory Standards (3)
Licensing (13)
RO Files

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RO:HQ (4)
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RO Files

B. RO Inquiry Report No. ______

Transmittal Date : ______

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C. Incident Notification From: ______
(Licensee & Docket No. (or License No.)

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