U. S. ATOMIC ENERGY COMMISSION
DIRECTORATE OF REGUALTORY OPERATIONS

REGION III

RO Inspection Report No. 050-263/73-04

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 27-29 and April 4-5, 1973

Dates of Previous Inspection: February 27 - March 1, 1973

Prinicpal Inspector: H C Dance for P. H. Johnson — 5/10/73 (Date)

Accompanying Inspectors: H C Dance H. C. Dance (March 28-29 only) — 5/10/73 (Date)

C. C. Williams (March 27 only)
Engineering Inspector

Other Accompanying Personnel: None

Reviewed By: H C Dance H. C. Dance, Senior Reactor Inspector
Reactor Operations Branch — 5/10/73 (Date)

SUMMARY OF FINDINGS

F[illegible]t Action

[illegible]pecification 3.7.A.2. states that primary containment integrity [illegible] maintained when the reactor water temperature is above 212°F and fuel is in the reactor vessel.

Contrary to the above requirement, primary containment integrity was not in effect on March 3, 1973, when a torus manway cover was removed. (Paragraph 10)

Licensee Action on Previously Identified Enforcement Matters

The licensee has completed corrective actions related to noncompliance items 9.a., 9.d., 9.i. and 14, as noted during the May 1972 management inspection. (Paragraphs 4, 14, and 15) Corrective action related to item 15 is continuing. (Paragraph 5) Corrective actions on remaining items were not reviewed during this inspection.

Unusual Occurrences

The ball valve associated with the No. 1 traveling in-core probe failed to isolate on February 2, 1973. (Paragraph 11)

Other Significant Findings

A. Current Findings

1. Unresolved Item: A clarification of code requirements relating to overpressure protection for off-gas storage tanks will be obtained. (Paragraph 18)

2. In-service inspection to the requirements of ASME Section XI, 1971, was conducted by the licensee durin[illegible] this major outage, in areas specified in Table 4.6.1 of the [illegible]ical specifications. A review of the QA/QC provisions and procedures associated with the in-service inspection program indicated conformance in all areas. (Paragraph 20)

3. Regarding the valve wall thickness verification program, the licensee reported that he plans to examine all valves identified to be within the area of concern during the current outage. Approximately 80 such valves have been identified. Followup review is planned for a future inspection. (Paragraph 22)

Management Interviews

Three separate management interviews were conducted during the course of the inspection. Mr. Williams conducted an interview on March 27, after completing a review of the licensee's in-service inspection and valve wall thickness verification programs. The interview was attended by Messrs. Larson (Plant Manager), Krumpos (Acting Site QA Engineer), Lambert (Senior Consultant Engineer), and Johnson (RO:III). The following matters were discussed:

A. The inspector said that his review of QA/QC considerations and procedures related to in-service inspection activities indicated that the examination program appears to conform to the requirements. However, the inspector requested that a final copy of the site proposed and implemented addenda to the UT procedures regarding "transfer" be made available to RO:III.

   The licensee acknowledged the inspector's remarks regarding the acceptability of their in-service inspection program and stated that a final copy of the UT transfer addenda would be available within two days. (The addenda was subsequently provided and appeared to meet requirements of ASME Section XI).

B. The inspector stated that his review of the valve wall thickness verification program and its implementation indicated that the technological parameters of the examination procedures, including the number of measurements, per valve, appears adequate. The licensee stated that all valves requiring wall thickness verification would be examined during the current reactor outage. The licensee added that approximately 72 valves have been examined so far and that the expected total would exceed 80.

Messrs. Dance and Johnson conducted a management interview with Mr. Larson (Plant Manager), on March 29, 1973, at which the following topics were discussed:

A. The inspector stated that entering the torus before reactor temperature was reduced below the minimum temperature requiring containment integrity represented an item of noncompliance. He stated that the letter of notification would not require a written response, in view of the corrective actions outlined by the licensee in his report to the Directorate of Licensing. (Paragraph 10)

B. The inspector noted that development of the plant's operating quality assurance program was progressing steadily but was some period of time behind the schedule objective established in the licensee's response to the management audit enforcement letter. (Paragraph 5)

C. The inspector stated that the licensee's corrective actions related to items 9.a., 9.d., and 9.i. and 14 of the management audit enforcement letter had been reviewed and were considered to have been completed. (Paragraphs 4, 14, and 15)

D. The inspector stated that during a tour of the drywell it had been noted that the B safety valve discharge was directed at portions of the A relief valve, particularly the conduit which carries the electrical conductor associated with the automatic depressurization system. The licensee stated that nozzle orientation would be reviewed for optimum positioning.

Mr. Johnson conducted a management interview with Mr. Larson (Plant Manager) on April 5 at the conclusion of the inspection, at which the following additional matters were discussed:

A. The inspector stated that overpressure protection requirements for offgas storage tanks had been discussed with cognizant individuals and that the matter was considered to be an unresolved item pending clarification of the code requirements. (Paragraph 18)

B. The licensee stated that the affixing of warning signs to drywell and torus sensing lines would be completed prior to the end of the outage. The inspector stated that he would review the documentation of the completion of corrective action related to this item during the next inspection.

C. The licensee stated that review of the semi-annual report for the period ending December 31, 1972, had led to a determination that the amount of solid waste reported to have been shipped from the plant during the period was in error. The amount reported was approximately 100 times the actual amount shipped, as a result of a clerical error in the typing of the report. He stated that a correction would be submitted to the Directorate of Licensing.

D. The inspector noted that an internal memorandum had reported the use of common connection boxes, conduit, and power sources by redundant logic channels of the HPCI and RCIC steam line isolation circuitry. The licensee stated that the matter was still under review and that the need for examination of other systems was being investigated. (Paragraph 16)

REPORT DETAILS

1. Persons Contacted

Monticello Plant Staff

C. Larson, Plant Manager
M. Clarity, Superintendent - Plant Engineering and Radiation Protection
W. Anderson, Superintendent - Operations and Maintenance
G. Jacobson, Plant Engineer, Technical
M. Dinville, Plant Engineer, Operations
L. Eliason, Radiation Protection Engineer
P. Krumpos, Acting Site Quality Assurance Engineer
D. Nevinski, Nuclear Engineer
M. Hammer, Engineer
D. Antony, Engineer
W. Shamla, Instruments Engineer
L. Nolan, Engineer
R. Rogers, Lead Plant Equipment Operator

Modified Off-Gas System Project

J. Meier, NSP Quality Assurance Engineer
J. Sevier, Test Director, Suntac Nuclear Corporation

Nuclear Services Corporation (NSC)

T. G. Lambert, Senior Consultant Engineer - NDT

2. General

At the time of this inspection, the Monticello plant was shut down for a refueling outage which began in early March and which is expected to be completed in late May 1973.

3. Record Reviews

The following records were reviewed without comment during the inspection:

a. Operations Committee Minutes for meetings conducted February 13, 19, 20, 21, 23, and 28, and March 1, 12, 18, and 20, 1973.

b. Safety Audit Committee minutes for meetings conducted February 14-15, 1973. Review by the Committee of the power range monitor decalibration event[1/] of May 1973, was noted to have been included.

1/ RO Inspection Report No. 050-263/73-01.

c. Reactor and Control Room Log, February 15 - March 15, 1973.

4. Test Documentation

Item 9, Part d, of the RO:HQ enforcement letter[2/] following the 1972 management audit noted that test results required by surveillance test procedure No. 0004 had not been recorded for tests conducted in August and October, 1972. The licensee's reply[3/] stated that new instructions had been issued to the instrument crew and that a revised procedure format would improve the documentation of testing results. The inspector reviewed all completed test documents for procedures 0004 since January 1972, and found all required data to have been recorded. The memorandum to instrument personnel referred to in the licensee's letter was also examined. A previous inspection report[4/] discusses the improvements in surveillance test procedure format made by the licensee following the management audit. Based on the foregoing, the inspector considered the licensee's corrective action on item 9d to have been completed.

5. Quality Assurance Program

The inspection included a discussion with persons responsible for formulation and issue of the facility's Quality Assurance Program. The licensee's response[5/] to the RO:HQ enforcement letter following the management audit discussed the licensee's plans regarding the establishment of a formal Quality Assurance Program. The response stated that the program will include a policy manual, to be issued by the General Superintendent - Nuclear Power Plant Operation, and an implementing program of Administrative Control Directives, to be issued by the Plant Manager. The inspector obtained a copy of the Administrative Controls Manual for review. It was noted that the manual has been expanded slightly to now include a total of 71 separate Administrative Control Directives. Of these, three have been issued by the Plant Manager, one additional directive has been issued for trial use and comment, four others were stated to be almost ready for issue, and an additional three were undergoing final review. A total of 41 additional directives were given to the inspector in the form of second drafts, for a total of 52 of the 71 directives in draft or final form. The inspector noted that implementation of the program was a few months behind the scheduled objectives set forth in the licensee's response,[6/] and that no other

2/ Letter, RO:HQ to NSP, dated 10-19-72.
3/ Letter, NSP to RO:HQ, dated 11-10-72.
4/ RO Inspection Report No. 050-263/72-07, p.9.
5/ Letter, NSP to RO:HQ, dated 11-10-72.
6/ Ibid.

schedule appeared to have been established. Licensee representatives did not differ with the inspector's observation. The inspector stated that the Quality Assurance Program would be discussed in more detail during a future inspection after review of the material provided.

6. Barton Differential Pressure Switches

The inspector asked whether Barton differential pressure switches of models 368, 384, or 386 were in use at the Monticello facility, in that these had been reported to be problematic at other facilities. A licensee representative stated, after checking a facility listing, that switches of this model and type were not in use at Monticello.

7. Control Rod Drive (CRD) Accumulators

During the inspection the inspector inquired about the operating history of CRD accumulators. A facility representative stated that three accumulator upper end seal O-rings had been replaced since the plant commenced operation. Two cases of minor plating deterioration had also been noted, both having been indicated by water level alarms in the lower portion of the accumulator resulting from scoring of the piston O-rings. Of two maintenance documents reviewed by the inspector relating to the CRD accumulators, one reported an area of deteriorated plating approximately 1/4" near the upper end seal O-ring. The representative stated that routine inspection of accumulator internals was not currently envisioned, since deterioration of the plating would be expected to show up as a piston seal leak.

The licensee representative showed the inspector a letter from General Electric Company, dated February 28, 1973, which discussed the possibility of accumulator plating deterioration. It stated in part that accumulator performance is monitored by gas pressure and water level switches, which indicate in the control room, and that if leaking did occur, scram performance should not be affected due to the force margins involved and the thickness (0.0015") of the plating.

8. Main Steam Tunnel Temperature Switch

A licensee report[7/] discussed the discovery during surveillance testing that one of the 16 "main steam tunnel high area temperature-group I isolation" temperature switches was found to have a trip set point of 204°F, which is four degrees above the Technical

7/ Letter, NSP to Directorate of Licensing, dated 3-19-73.

Specification limit. The report, which described the licensee's corrective action, also stated that redundant temperature switches would have provided the desired trip action had isolation been required. The inspector reviewed the licensee's significant operating event report and raised no further questions.

9. Uninterruptible AC Power System

A previous inspection report[8/] discussed operating difficulties experienced by the licensee with the AC-DC motor-generator used in the uninterruptible AC power system. The report also noted, as stated in the FSAR, that the high pressure coolant injection system was dependent upon this uninterruptible system for operation. A licensee representative stated during the current inspection that he had since determined the HPCI flow controller power supply had been modified during the startup testing program to receive power from either of two essential sources, each of which is backed up by a diesel-generator. The inspector verified this to be the existing power supply arrangement by reviewing current systems schematic diagrams. A licensee representative also stated that following rerouting of the tachometer cable associated with the motor generator and the replacement of a silicone control rectifier in its control system no further problems had been encountered, and that the unit had been operating smoothly for approximately one month.

10. Violation of Primary Containment

A licensee report[9/] to the Directorate of Licensing stated that on March 3, 1973, a torus manway was opened with reactor temperature at approximately 240°F, in violation of Technical Specification 3.7.A.2 requirement that primary containment integrity be maintained above a reactor temperature of 212°F. The inspector confirmed that administrative instructions had been issued to the staff in regards to the precautions required to include recurrence. The posting of the caution signs was scheduled to be completed during the current outage.

11. Traveling In-Core Probe (TIP) Ball Valve

A licensee report[10/] discussed the failure of a TIP ball valve to isolate on February 2, 1973, after retraction of its associated TIP (The ball valve is opened by a solenoid and closes by spring tension) Review of the licensee's significant operating event report and

8/ RO Inspection Report No. 050-263/72-06.
9/ Letter, NSP to Directorate of Licensing, dated 3-12-73.
10/ Letter, NSP to Directorate of Licensing, dated 2-23-73.

discussion with a licensee representative indicated the event to have been as described in the licensee's report. The inspector examined a Volume F temporary memorandum requiring verification of ball valve closure on a daily frequency when containment integrity is required.

12. Primary Containment Isolation Valve Leakage

A previous inspection report[11/] discussed leakage experienced with primary containment isolation valves (including torus to reactor building vacuum breakers) which use a pneumatically pressurized rubber seal. Licensee representatives stated that disassembly and inspection of the valves gave reason to question the dimensions of the rubber "T ring" seals (so named because of the cross sectional shape), and that the assistance of a manufacturer's representative had been requested. Based upon his recommendations, the inner diameter of all T-rings was increased by 0.020 inches, and a ridge was removed from the T-ring retaining slot in the valve cover on one valve. The licensee expected these improvements and the application of a lubricant to the T-rings to provide proper operation.

13. Torus-Drywell Vacuum Breakers

A previous inspection report[12/] summarized the licensee's plan to improve operation of torus-drywell vacuum breakers. The modifications were observed to be in progress during a tour of the torus by the inspector. The rubber and metal seating surfaces were noted to be free of foreign material and the vacuum breakers were observed to be freely operable with the manual actuating arm. The installation of new microswitches had been completed on most of the vacuum breakers. The inspector noted by examination of those vacuum breakers on which the modification had been completed that the new switches are sensitive to disc movement of approximately 1/16" from the fully closed position. A licensee representative stated that the new microswitches were also of a design which would make them more resistant to the humid atmosphere inside the torus. The inspector verified by examination of records that the modification had been reviewed in accordance with 10 CFR 50.59. It was approved by the Operations Committee on February 28, 1973.

14. Drywell Leak Rate Monitoring

A previous inspection report[13/] discussed actions taken by the licensee in response to item 9, parts a and i, of the RO:HQ enforcement letter[14/] following the 1972 management audit. A licensee representative stated

11/ RO Inspection Report No. 050-263/73-01.
12/ RO Inspection Report No. 050-263/72-03.
13/ RO Inspection Report No. 050-263/72-02.
14/ Letter, RO:HQ to NSP dated 10-19-72.

that investigation into difficulties with operation of the float switches which start and stop the equipment and floor drain sump pumps led to a determination that the original switches were not suitable for use in humid areas. He stated that purchase of new switches more suitable for use in the humid conditions was being studied. He aslo stated that the float balls and stems had appeared to be in good condition, that system operation prior to the recent switch problem had been smooth, and that satisfactory operation would be expected after replacement switches were installed. The inspector reviewed new procedures which had been issued to cover "FLOOR DRAIN LEAK RATE CHANGE HIGH" and "EQUIPMENT DRAIN LEAK RATE CHANGE HIGH" alarm action. These call for the sump pump timers to be set daily at a pump run time interval which will correspond to a leak rate of 0.2 gpm greater than the previous 24-hour average. Based on the foregoing and upon the information provided in the earlier inspection referenced above, the inspector considered the licensee's corrective actions on items 9a and 9i of the RO:HQ enforcement letter to be complete.

15. Semi-Annual Report

The semi-annual operating report for the period ending December 31, 1972, was submitted as required by Section 6.6.D of the Technical Specifications. Review of the report indicated the reporting format in Safety Guides 16 and 21 had been followed as requested by a letter from the Directorate of Licensing to the licensee dated November 20, 1972.

In addition, the above referenced semi-annual report contained a listing of facility changes prior to March 1971. This matter was the subject of Item 14 of a Regulatory Operations enforcement letter dated October 19, 1972, and the licensee's response dated November 10, 1972. The inspector's review noted that the Operations Committee meetings minutes dated March 3, 1971, recognized requirements of 50.59 and discussed means for updating. The inspector confirmed that March 1971 in-plant reports from staff engineers existed for systems comparison of the as-built plant to that discussed in the FSAR. Discussion with two staff engineers indicated that a review of Work Request records in addition to discussion with responsible staff members was the principal basis for this review. The inspector considered the licensee's corrective actions related to item 14, as discussed in his letter of November 10, 1972, to be complete.

The inspector also noted that facility changes for 1973, No. 91 through No. 98, had been or were being formally reviewed for 50.59 requirements. Recent addenda to facility changes No. 97 and No. 98

had not been reviewed at the time of the inspection. The present administrative control policies appear adequate to assure reporting of 50.59 plant modifications.

16. Safeguards System Circuitry

During review of significant operating event reports, the inspector noted a March 5, 1973, memorandum which stated that a review (prompted by a General Electric Company request) had determined that HPCI and RCIC high steam flow isolation circuitry uses common conduit, connection boxes, and power sources for redundant logic channels. The memorandum noted that a General Electric Field Design Instruction had been issued and that a Field Design Change Notice was pending to correct the installation. During the final interview, the inspector asked whether the licensee planned a review of related systems for similar conditions. A management representative stated that review of other diagrams had led to the conclusion that the primary containment group isolation circuitry was properly separated, but that the need for further examination of other systems was being investigated.

17. Procedures for Drywell Entry

A previous report[15/] discussed a small increase in airborne radioactivity level within the drywell while maintenance was being performed. The licensee had concluded that gaseous activity had been released from the reactor head vent. The inspector noted that a recent revision to Section B.4.1 of the Operations Manual requires the head vent to be tagged in the closed position prior to drywell entry.

18. Off-Gas System Modification

A representative of plant management stated during the inspection that, partly due to delay caused by a brief strike during the month of April, it appeared that the modified off-gas system would not be ready for operation prior to plant startup at the completion of the current outage. He stated that plant connections were being installed, but that these would be blanked off until system construction and testing are completed. Final connection and checkout of the system will be accomplished during a future outage which the licensee hopes will be scheduled in early summer, 1973.

The inspector examined during the inspection a memorandum reporting the results of an inspection of off-gas system storage tanks conducted by a licensee quality assurance representative on January 23, 1973,

15/ RO Inspection Report No. 050-263/72-06.

in response to an intention expressed during a previous inspection[16/]. The memorandum stated that the tank had been inspected and found acceptable. The dessicant was removed, the tanks were brushed and vacuumed, and the connecting nozzles were blown out with compressed air. The inspector noted during a tour of the off-gas system that relief valves were installed only on the discharge of the compressors, the only route for tank pressurization, and not individually on each of the five isolable storage tanks. The inspector noted that Article IX of the 1968 ASME Code, Section III, to which the storage tanks were built, appeared to require separate overpressure protection for the individual storage tanks. Licensee representatives stated that their interpretation, based partly on amplifying statements in the corresponding section of the 1971 ASME Code, was that separate overpressure protection was not required, but that a clarification of the requirement would be obtained. The inspector stated that this was considered to represent an unresolved item pending such clarification.

19. Refueling

This inspection included review and examination of the licensee's refueling activities, which were in progress at the time of the inspection. The inspector verified that the following technical specification requirements had been complied with prior to and during refueling operations:

a. Reactor mode switch locked in "refuel" position

b. Test of refueling interlocks conducted prior to fuel handling and weekly when handling operations in progress

c. Source range monitor functional test conducted prior to core alterations and daily (March 11-29) when refueling operations in progress

d. Fuel pool level recorded daily when irradiated fuel located in fuel storage pool (verified for the period March 15-28)

The inspection also included observation, without comment, of the following activities:

a. Fuel removal and insertion

b. Poison curtain removal

16/ RO Inspection Report No. 050-263/72-03.

c. Use and signoff of refueling procedures

d. Refueling communications (use of proper acknowledgements)

e. Reconstitution work in progress

f. Fuel accountability (including spot check of status boards)

g. Status keeping and procedure following in control room

h. Supervision of handling activities by senior licensed operator with no concurrent duties

The inspector also viewed selected portions of videotapes showing (1) checking of jet pump tack welds, (2) visual inspection of CRD housing to stub tube welds and portions of the inner surface of the lower reactor vessel head, and (3) examination of a portion of the upper core grid. Fuel reconstitution was essentially complete at the conclusion of the inspection. A licensee representative stated that of 1248 fuel rods inspected (1 good bundle and 25 potentially leaking bundles, as identified by in-place sipping), a total of 163 had been rejected, although only 41 of these had given both positive eddy current and ultrasonic test indications. A positive eddy current test was considered to be an indicated reduction in wall thickness of 45 percent of more. A total of 24 fuel rods gave indications (100 percent eddy current indication) of having had fuel clad perforations. The off-gas release rate was stated to have been approximately 40,000 uCi/sec prior to commencement of the outage, at which time the average core exposure was 7115 MWD/T. The licensee representative stated that existing plans called for plant startup with 20 new fuel assemblies and 44 control curtains as compared to the 28 assemblies and 64 curtains described in the licensee's submittal to Licensing of February 20, 1973. He stated that Licensing would also be notified of the revised loading plan (subsequently accomplished by an NSP letter dated April 13, 1973).

20. In-service Inspection - Reactor Coolant System

The inspection of the in-service inspection program included a review and examination of all applicable QA/QC provisions, nondestructive test procedures, and "technical specificaton requirements". The examinations were performed pursuant to the requirements of ASME Section XI, "Rules for In-service Inspection of Nuclear Reactor Coolant Systems". NSP had defined the scope of their commitment to ASME Section XI in the "Monticello Technical Specifications", Table 4.6.1, titled, "In-service Inspection Requirements for Monticello". Although changes to this document are in progress (to reflect the latest revision of ASME Section XI), applicable aspects of the revisions were included in the current examination.

The Monticello Operation Quality Assurance Program is currently in draft form and, therefore, its provisions were not considered relative to the current phase of the examination program. However, the inspection did establish that the appropriate aspects of 10 CFR Part 50, Appendix B, and ASME Section XI (1971) were met in the conduct of these examinations.

The examinations were conducted under the direction of the site quality assurance engineer, who holds a Level III ASNTC certification. Tests and examinations were performed by NSC personnel, under the direction of an NSC senior consultant for NDT examination. The assigned code inspectors reviewed each procedure, witnessed each test, and signed the data sheets. A Hartford Insurance code inspector had previously reviewed and concurred with the various procedures.

21. Details of Inspection

a. Review of the Monticello Technical Specification (TS) Requirements - Table 4.6.1

The TS Table 4.6.1 established the following:

(1) Examination categories, areas, extent, and frequency are in accordance with Table IS-251 of ASME Section XI.

(2) Examination methods are in accordance with Table IS-261 of ASME Section XI.

(3) The extent of examination is in terms of accessibility of the components.

TS Table No. 4.6.1 is presently undergoing revision in order to reflect the latest revisions of ASME Section XI.

The portion of the required ten-year examination program accomplished during this reactor outage appears to have met the intent of the requirements of Section XI subject to the restrictions of component accessibility.

b. Examination Techniques and Procedures

All nondestructive test procedures were provided by NSC.

(1) Visual Examination

Special process standard No. NVT-NC-1, reviewed and approved by NSP QA, the assigned code inspector, and the vendor's representative, was determined to be conforming to all requirements.

(2) Surface Examination

Nuclear penetrant testing procedure No. NPT-NC-1, Revision 1, approved by NSP QA, the code inspector, and the vendor's representative, was found to be in conformance to the requirements.

(3) Volumetric Examination

(a) "Nuclear Ultrasonic Testing Procedure for Ultrasonic Inspection of Full Penetration Butt and Nozzle Welds" (NUT-NC-1A) Revision 0, was reviewed and approved for use by the NSP representatives. The inspector found the procedure to be conforming to requirements of Section XI with the exception that it did not clearly relate to the ultrasonic "transfer" considerations of ASME Section III. However, it was determined that the "transfer" considerations (although undocumented) were implemented during these examinations. This oversight was corrected by an addenda to the subject procedure during the inspection. Ultrasonic tests, which occurred previous to writing of the addenda, were not affected. The addendu dealt only with the requirement to clearly document the procedures and considerations which were actually implemented for these tests.

(b) Nuclear Ultrasonic Testing Procedure - "Longitudinal Wave Ultrasonic Inspection of Vessel Flange, Ligaments, Vessel Studs, and Nuts" (No. NUT-NC-1B) was observed to have been properly approved. The inspector determined that the procedure met requirements except that UT transfer considerations were not clearly stipulated. This condition was immediately corrected by an addenda to the procedure (as in (a) above).

c. Qualification of Nondestructive Examination Personnel

Review of the test vendor's personnel qualification documentation indicated conformance to the requirements of SNT-TC-1A, its supplements, and appendices as applicable for the examination technique and methods used. Full resumes, written tests and results, employment and experience history, were available for each member of the crew. Additionally, the status, criteria, and results of vision examinations were available. All documents had been reviewed by NSP QA and the assigned code inspector.

d. Basic Calibration Status of Instruments

(1) Electronic

NSC provided complete and comprehensive records attesting to the establishment and maintenance of a calibration program to measure and verify the accuracy of all measuring and test equipment. Each UT instrument observed had a calibration tag attached with the calibration status noted.

(2) Transducers

NSC provided documentation attesting to the conformance of each transducer to the qualitative and quantitative requirements of the ASME code, specifications, and/or test procedures. Certifications and actual test data were available, and the records had been reviewed by NSP QA.

e. Material Certifications

(1) Material certification for all liquid penetrant materials were reviewed by the inspector and confirmed as applying to the materials used.

(2) Material certifications for the ultrasonic test couplant were available and were found to conform to the requirements of the procedure.

(3) Material certifications for each UT calibration reference block were reviewed and found to establish the acceptability of each block. Moreover, each reference standard was determined to be adequately traceable to the certification. The surface condition of the reference standards were judged to be representative of the as-built surface condition of the test specimens by the code representative and NSP QA.

(4) Material certification relating to the "paint sticks" used to mark the inspection surface, were available and were found to conform to requirements.

f. Test Records

The inspector reviewed reports of test results and found that requirements of ASME Section XI regarding documentation of the examinations had been met or exceeded. Each data sheet was signed by the vendor's representative, the assigned code inspector, and NSP QA. Data acquisition was taken manually and by automatic (annolog) printer, as appropriate for the examination method.

22. Valve Wall Thickness Verification Program

The inspector reviewed the QA/QC considerations and procedures for the valve wall thickness verification program. The basis for this review was the commitment established in the NSP letter dated February 20, 1973, in response to RO:III letter dated January 22, 1973. RO:III acceptance of the valve wall thickness verification program, proposed by NSP, had not been established at the time of the current inspection. However, subsequently the program commitments contained in the NSP letter of February 20, 1973, have been accepted.

The NSP proposal identified 72 valves which would be examined for adequate wall thickness and stated that an ultrasonic procedure for this examination would be developed. NSP had earlier stated that only 32 of the identified valves would be examined in 1973.

During this inspection, the licensee's representative reported that the examination procedure has been developed, approved, and implemented. NSP reported that approximately 72 of the valves have been examined and that they expect that the final number of valves examined will exceed 80. It is their (NSP's) intent to examine all valves so identified during the current outage, although the RO:III AEC letter of June 29, 1972, would allow up to three years to accomplish this examination.

a. Review of the valve wall thickness ultrasonic measurement procedure indicated that the technical approach and coverage is comprehensive and appropriate. The following details were noted:

(1) The procedure (No. NUT-NC-2, Revision 2) was developed by the NSP consultant, NSC, has been approved by NSP QA, and reviewed by the assigned code inspector.

(2) It provides for comprehensive calibration of the UT instrumentation with calibration blocks of cast and forged materials as appropriate, each of which conform to the P group requirements of the code.

(3) The grid pattern for measurements is as follows:

(a) For valve diameters less than six inches, a two-inch grid will be used.

(b) For valve diameters between two and six inches, a minimum of twelve readings will be provided.

(c) Valves six to fourteen inches in diameter shall be examined on a three-inch grid.

(d) Valves larger than fourteen inches shall be examined on a six-inch grid.

b. Review of the rough records from valve wall thickness verification examinations, disclosed the following:

(1) No significant deviation from requirements have been identified.

(2) The data sheets include the code specified thickness requirements, identification of the foundry supplier, the UT instrument used to make the measurements, and sign off by NSP QA engineering.

Further review of this matter is planned for future inspections.

UNITED STATES
ATOMIC ENERGY COMMISSION
DIVISION OF COMPLIANCE
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

TELEPHONE
(312) 858-2660

REGULATORY OPERATIONS, REGION III

A. RO Inspection Report No. 050-263/73-04

Transmittal Date : May 11,1973

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

Distribution:
RO Chief, FS&EB
RO:HQ (4)
L:D/D for Fuel & Materials
DR Central Files

B. RO Inquiry Report No. _____

Transmittal Date : _____

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

Distribution:
RO Chief, FS&EB
RO:HQ
DR Central Files

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

Transmittal Date : _____

Distribution:
RO Chief, FS&EB
RO:HQ (4)
Licensing (4)
DR Central Files

Distribution:
RO Chief, FS&EB
RO:HQ (4)
L:D/D for Fuel & Materials
DR Central Files