940111DRP1

November 28, 1994

Northern States Power Company
ATTN: Mr. E. Watzl
Vice President, Nuclear
Generation
414 Nicollet Mall
Minneapolis, MN 55401

Dear Mr. Watzl:

This refers to the routine safety inspection conducted by Messrs. S. Ray, W. Stearns, J. Gavula, D. Butler, and S. Orth of this office as well as Ms. B. Wetzel of the Office of Nuclear Reactor Regulation from September 27 through November 15, 1994. The inspection included a review of activities at the Monticello Nuclear Generating Plant authorized by NRC Operating License No. DPR-22. At the conclusion of the inspection, the findings were discussed with those members of your staff identified in the enclosed report.

Areas examined during the inspection are identified in the report. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations, and interviews with personnel. The purpose of the inspection was to determine whether activities authorized by the license were conducted safely and in accordance with NRC requirements.

Based on the results of this inspection, certain of your activities appeared to be in violation of NRC requirements, as specified in the enclosed Notice of Violation (Notice). We remain concerned about the number of events which involve inadequate attention to procedures and failure to self-check work. As discussed with your staff, two events discussed in the report are considered additional examples of the Notice of Violation that accompanied Inspection Report No. 50-263/94009. Your staff agreed to include corrective actions for these two events in the response directed by that report.

Other activities discussed in this report also appeared to be in violation of NRC requirements. However, the violations were not cited in accordance with Section VII.B of Appendix C to 10 CFR Part 2, the NRC's Enforcement Policy, because they were of minor safety significance and adequate corrective actions were taken or were underway by the conclusion of the inspection.

We did note strengths in your operations during the inspection period. Most notably the scheduling, teamwork, and risk control displayed during the recent refueling outage was considered excellent.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. In your response, you should document the specific actions taken and any additional actions you plan to prevent recurrence. After reviewing your response to this Notice, including your proposed corrective actions and the results of future inspections, the NRC will determine whether further NRC enforcement action is necessary to ensure compliance with NRC regulatory requirements.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, the enclosures, and your response to this letter will be placed in the NRC Public Document Room.

The response directed by this letter and the accompanying Notice are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, Pub. L. No. 96-511.

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

Sincerely,

Original signed by
Edward G. Greenman, Director
Edward G. Greenman, Director
Division of Reactor Projects

Docket No. 50-263

Enclosures:
1. Notice of Violation
2. Inspection Report
   No. 50-263/94011(DRP)

cc w/encl: Site General Manager, MNGP
W. Hill, Plant Manager
John W. Ferman, Ph.D.,
  Nuclear Engineer, MPCA
State Liaison Officer, State
  of Minnesota

Distribution:
Docket File w/encl
PUBLIC IE-01 w/encl
OC/LFDCB w/encl
SRI, Monticello,
  Prairie Island w/encl
DRP w/encl
RIII PRR w/encl
Project Manager, NRR w/encl

Document: R:\insprpts\powers\mont\mon94011.drp
To receive a copy of this document, indicate in the box "C" = Copy without attach/encl "E" = Copy with attach/encl "N" = No copy

OFFICE	RIII	E	RIII	E	RIII	C	RIII	C	RIII		RIII	
NAME	Harper		Phillips		Jacobsen		JMC-B*		Pirok		Greenman	
DATE	11/23/94		11/25/94		11/25/94		11/23/94		11/25/94		11/[illegible]/94	

OFFICIAL RECORD COPY

* Also obtained NRR concurrence on violations.

NOTICE OF VIOLATION

Northern States Power Company
Monticello Site

Docket No. 50-263
License No. DPR-22

During an NRC inspection conducted on September 27 through November 15, 1994, violations of NRC requirements were identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C, the violations are listed below:

1. Technical Specification 6.5.C.3 requires, in part, that detailed written procedures, including applicable check-off lists and instructions, covering maintenance and test procedures for preventive or corrective maintenance of plant equipment and systems that could have an effect on nuclear safety. Work Request Authorization 94-05322 provided instructions for the repair of tack welds on adjusting screws located on the reactor vessel jet pumps.

   a. Attachment 2 to WRA 94-05322 was GE instruction FDI 0382-51847. Step 4.2 of that instruction required that a functional test of the welding procedure and equipment be performed with the welding fixture submerged in water. This step also required the completion of three acceptable test tack welds in accordance with the qualified welding procedure specification and that these welds be broken by a torque test (25 ft-lb).

      Contrary to the above, on October 10, 1994, the weld machine and procedure for welding had not been functionally tested prior to use on jet pump set screw tack welds.

   b. Attachment 3 to WRA 94-05322, GE procedure 25A5589 rev. 0, "Underwater Tack Welding," required in step 3.4.1 that all production welds shall be performed in accordance with the qualified weld procedure. Additionally the step required that no welding shall be performed using parameters outside the range of the qualified essential variable without requalification of the procedure.

      Contrary to the above, on October 10, 1994, welds were completed with the welding machines' essential variables set to values outside of their qualified ranges.

   c. Attachment 3 to WRA 94-05322, GE procedure 25A5589 rev. 0, "Underwater Tack Welding," required in step 4.b that welding settings, including current, voltage, time, and downslope time as a minimum be recorded and maintained.

      Contrary to the above, on October 10, 1994, welding settings used to make the welds were not recorded for seven welds.

   This is a Severity Level IV Violation (Supplement I).

2. 10 CFR 20.1501 requires that each licensee make or cause to be made surveys that may be necessary for the licensee to comply with the regulations in Part 20 and that are reasonable under the circumstances to evaluate the extent of radiation levels, concentrations or quantities of radioactive materials, and the potential radiological hazards that could be present.

   Pursuant to 10 CFR 20.1003, *survey* means an evaluation of the radiological conditions and potential hazards incident to the production, use, transfer, release, disposal, or presence of radioactive material or other sources of radiation.

   Contrary to the above:

   a. During the September 1994 outboard main steam isolation valve (MSIV) replacement, the licensee did not make adequate surveys to assure compliance with 10 CFR 20.1201 which limits occupational radiation exposure to an adult. Specifically, the licensee failed to identify and evaluate the alpha radioactivity component in the radiological planning for the evolution.

   b. During the September 1994 inboard MSIV replacement, the licensee did not make surveys to assure compliance with 10 CFR 20.1201 which limits occupational radiation exposure to an adult. Specifically, the licensee failed to provide an adequate airborne radioactivity evaluation prior to allowing entrance into the drywell MSIV area without respiratory protection.

   This is a Severity Level IV violation (Supplement IV).

Pursuant to the provisions of 10 CFR 2.201, Northern States Power Company is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Regional Administrator, Region III, and a copy to the NRC Resident Inspector at the Monticello Nuclear Generating Plant, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. If an adequate reply is not received within the time specified in this Notice, an order may be issued to show cause why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

Dated at Lisle, Illinois
this 28 day of November 1994

NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-263/94011(DRP)

Docket No. 50-263 License No. DPR-22

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

Facility Name: Monticello Nuclear Generating Plant

Inspection At: Monticello Site, Monticello, MN

Inspection Conducted: September 27 through November 15, 1994

Inspectors: S. Ray W. Stearns
J. Gavula D. Butler
S. Orth B. Wetzel

Approved By: M. P. Phillips, Chief
Reactor Projects Section 2B

11/25/94
Date

<u>Inspection Summary</u>:
<u>Inspection on September 27 through November 15, 1994</u>
<u>(Report No. 50-263/94011(DRP))</u>

<u>Areas Inspected</u>: A routine, unannounced inspection by the resident inspectors and others of operations; maintenance; engineering; and plant support activities. The inspection also included inspections of radiation controls and a radiological sampling confirmatory measurements inspection.

<u>Results</u>: Two violations were cited in this report. One, with three examples, involved failures to follow procedures for welding activities in jet pump set screws (Section 3.1.3). The second, with two examples, involved inadequate contamination surveys for work on the main steam isolation valves (Section 5.1.3). In addition, two additional examples of violations previously cited involving failure to follow procedures were also identified (Sections 2.3 and 3.1.4). Several non-cited violations were also identified and are discussed below (Sections 2.4.2, 3.1.5, 4.1.4, 5.1.2, 5.1.4, and 5.1.).

The following is a summary of the licensee's performance during this inspection period:

<u>Operations</u>: Strengths were noted in control of refueling activities and in the conduct of physics testing and reactor startup. Communications, supervision, engineering assistance, and monitoring of indications, all of which have been identified as weaknesses in previous inspections, were considered good. This inspection period included a large number of

successfully completed major plant evolutions. The licensee's analysis of a misorientated fuel bundle was also considered good. An example of a violation was identified in that a licensed operator failed to properly follow a procedure for transferring reactor protection system power supplies and caused an engineered safety feature actuation. That event also involved inadequate operations supervision. Two non-cited violations were identified by the licensee concerning a fuel bundle that was mis-oriented during the previous cycle core reload, about one and one half years ago.

Maintenance: Communications, teamwork, risk control, and industrial safety were considered strengths during the refueling outage. The breadth and depth of quality services involvement in the outage work was considered strong. In addition, the timeliness of publishing and reviewing quality service findings continued to be excellent. Weaknesses were identified in procedure adherence during the jet pump tack weld repair and work on a valve in the steam chase. This resulted in one violation with three examples for failure to follow the welding procedure and one example of a violation for starting work on the wrong valve. Two non-cited violations were identified by the licensee for inadequate procedures in that two surveillance test procedures were not adequate to prevent unanticipated engineering safety feature actuations.

Engineering: No new weaknesses were identified. A strength was identified in the licensee's program of selection and design of reliability-based modifications in striving for continuous improvements in safety. Another strength was noted in the timeliness of system engineers' efforts in reviewing industry events and issues. A non-cited violation was identified by the licensee during a audit concerning a fire door that did not meet the appropriate code.

Plant Support: One violation was identified concerning the main steam isolation valve (MSIV) replacement evolution. The initial radiological evaluation failed to recognize the extent of alpha contamination in the corrosion film, and the licensee failed to provide an adequate survey prior to workers entrance into the drywell MSIV area following grinding. Additionally, two non-cited violations were identified by the licensee for inadequate engineering controls to limit airborne radioactive materials and inadequate procedures to ensure the use of respiratory protection equipment certified by the National Institute for Occupational Safety and Health/Mine Safety and Health Administration (NIOSH/MSHA). The licensee's self assessment of weaknesses during the replacement of snubbers was very thorough. One non-cited violation was identified for failure to immediately evacuate an area upon receipt of an electronic dosimeter alarm. The licensee's dose control and ALARA implementation continued to be very good, and performance in the NRC radiological confirmatory measurements was excellent. The radiological environmental monitoring program continued to be well implemented.

DETAILS

1.0 Persons Contacted

Northern States Power Company

L. Waldinger, General Manager Monticello Nuclear Site
M. Hammer, General Superintendent Maintenance
*W. Hill, Plant Manager
K. Jepson, Superintendent Chemistry and Environmental Protection
L. Nolan, General Superintendent Safety Assessment
M. Onnen, General Superintendent Operations
E. Reilly, Superintendent Plant Scheduling
C. Schibonski, General Superintendent Engineering
W. Shamla, Manager Quality Services
J. Windschill, General Superintendent Radiation Services

The inspectors also contacted other licensee employees including members of the maintenance, engineering, quality, and operating staffs.

*Denotes those attending the resident inspectors' exit meeting on November 15, 1994. Additional pre-exit meetings were held on October 10, 1994, and November 4, 1994, to discuss the findings in the radiation protection and confirmatory measurements portions of this inspection.

2.0 Operations

The plant was in a refueling outage from the beginning of the inspection period until October 23, 1994, when the outage ended as the main generator was put on line for a short period. Due to a problem with manual disconnects on the main transformer the plant had to be taken off line until October 24, 1994. The plant reached full power on October 26, 1994, and remained at power through the end of the inspection period.

2.1 Operational Safety Verification

The inspectors verified that the facility was being operated in conformance with the license and regulatory requirements and that the licensee's management was effectively carrying out its responsibilities for safe operation of the facility. The inspectors verified proper control room staffing and coordination of plant activities; verified operator adherence with procedures and technical specifications; monitored the control room for abnormalities; verified that electrical power was available; and observed the frequency of plant and control room visits by station managers. The inspectors also monitored various records, such as hold and secure card records, jumpers and bypasses, shift logs and surveillances, daily orders, and maintenance items.

Specific findings in this area are discussed in later sections of this report.

2.2 Onsite Followup of Events

During the 1994 refueling outage several events occurred which required notification to the NRC via the emergency notification system in accordance with 10 CFR 50.72. The events are listed below. For each event the inspectors verified that the licensee properly evaluated and reported the event in a timely manner and that immediate corrective actions were appropriate. More detailed discussions of specific events are contained elsewhere in this report. For each event the licensee either issued or intended to issue a licensee event report (LER) as a written followup.

2.2.1 Containment Isolation Valve Actuations: On October 13, the licensee reported that it had experienced a group II containment isolation of the shutdown cooling suction valves due to a feeder breaker for the division 2 uninterruptible power supply opening during an electrical switching evolution. Shutdown cooling was not in operation at the time and the event had little safety significance. Power was rapidly restored and all systems returned to normal. The licensee issued LER 94-013 as a written followup. No cause could be determined for the breaker opening. The LER is considered open pending completion of corrective actions.

2.2.2 Standby Gas Treatment System Actuation: On October 13, the licensee reported that it had experienced a momentary loss of power to the fuel pool and reactor building ventilation radiation monitors causing an actuation of the standby gas treatment system and reactor building ventilation isolation. In addition, containment sample isolation valves closed. The cause of the event was personnel error when the radiation monitor trips were not bypassed during an electrical switching operation. The systems were quickly restored and the event had little safety significance. The licensee issued LER 94-014 as a written followup. The LER is closed elsewhere in this report. This event is an example of a violation discussed later in this report.

2.2.3 Loss of Shutdown Cooling: On October 15, the licensee reported that it had experienced a group II containment isolation of the shutdown cooling suction valves while performing testing of the division 2 uninterruptible power supply. In this case shutdown cooling was in operation at the time and it was lost for approximately 30 minutes while systems were restored to normal. Reactor temperature increased approximately 1° fahrenheit during the time shutdown cooling was lost. Reactor temperature was low at the time and there was no significant chance of boiling. The cause of the event was later determined to be an electrical problem with a static switch logic card in the uninterruptible power supply. Although involving the same power supply, this event was not believed to be related to the October 13th event discussed above. The licensee issued LER 94-015 as a written followup. The LER is closed elsewhere in this report.

2.2.4 Valve Closure due to Inadequate Procedure: On October 17, the licensee reported that the shutdown cooling suction valves had automatically closed on a high pressure signal during backflushing of reactor pressure instrumentation lines. Shutdown cooling was secured at the time of the event in accordance with the test procedure but the procedure did not require the suction valves to be closed. Operators were about to close the suction valves at the time of the event in accordance with their normal shutdown cooling securing procedure. The event had little safety significance and was caused by an inadequate procedure as discussed later in this report. The licensee issued LER 94-016 as a written followup. The LER will remain open pending the inspectors' review and completion of appropriate corrective actions.

2.2.5 <u>High Pressure Coolant Injection (HPCI) System Isolation During Testing</u>: On October 23, the licensee reported that it had experienced a group IV containment isolation of the high pressure coolant injection (HPCI) turbine supply valves during HPCI operability testing. The valves had closed on a high steam line flow signal. After extensive troubleshooting and calculations, the licensee determined that the isolation would have been expected under the conditions during the test. The event had little safety significance and was caused by an inadequate procedure as discussed later in this report. The licensee intended to issue LER 94-017 as a written followup. The LER will remain open pending the inspectors' review and completion of appropriate corrective actions.

2.2.6 <u>HPCI Automatic Suction Switching from Condensate Storage Tank (CST) to Torus</u>: On October 26, the licensee reported that it experienced an automatic switching of the HPCI suction lineup from the CST to the torus during HPCI testing. The suction switch was caused by a high torus level because the HPCI turbine had been exhausting into the torus for longer than planned due to various problems encountered during the test. This caused both the torus temperature and level to increase toward values requiring operator action. Operators properly decided to solve the torus temperature problem first. That involved starting additional residual heat removal (RHR) pumps and additional RHR service water pumps. The turbine building operator was dispatched to the intake structure to check on the newly started RHR service water pumps. Operators then started the process of lining up to discharge torus water to the radwaste system. That involved locally closing breakers for motor operated valves in the turbine building. It took too long for the turbine building operator to get from the intake structure to the motor control center so operators were not able to complete the lineup quickly enough to prevent the high torus level signal. As part of the automatic transfer, the test return line isolated and the operators immediately secured the turbine. Thus no torus water was pumped to the CST and the event had little safety significance. The procedures used during the evolution were adequate and were being properly used. Although the event constituted a challenge to a safety system, it was not considered a violation. The licensee intended to issue LER 94-018 as a written followup. The LER will remain open pending the inspectors' review and completion of appropriate corrective action.

2.3 Failure to Follow Procedure Results in Automatic Closure of Containment Isolation Valves

As discussed above, on October 13, 1994, licensee personnel were performing Test 0379, "Electrical Protection Assembly (EPA) Functional Test." Part of the test involved transferring the reactor protection system (RPS) power supplies from their normal motor-generator sets to alternate AC sources.

Step 1 of Part A of Test 0379 required operators to transfer RPS Bus A to the alternate source using Operations Manual B.9.12-05, Special Procedure G.1. That procedure contained steps to bypass the A division of the reactor building plenum and fuel pool radiation monitors because a momentary loss of power to the division during the power supply transfer would cause an actuation of containment isolation logic.

When Part A of Test 0379 was completed, work continued on Part B. Step 45 required operators to transfer RPS Bus B to the alternate source using Operations Manual B.9.12-05, Special Procedure G.2. That procedure contained steps to bypass the other division of the reactor building plenum and fuel pool radiation monitors. The operator mistakenly assumed that the correct bypasses were already in effect from Part A of the procedure and proceed to transfer power. During the transfer the momentary loss of RPS power caused actuation of the containment isolation logic as expected.

The isolation caused an isolation of containment sample valves, reactor building ventilation isolation, and startup of the standby gas treatment system. All systems responded as expected and the signal was quickly reset and equipment returned to normal configuration. The event was caused by inadequate attention to detail and inadequate use of the procedure on the part of a control room operator. Inadequate control of the evolution by operations supervisors was also a contributing factor.

Technical Specification 6.5 required, in part, that detailed written procedures, including applicable check-off lists and instructions, covering the following, shall be prepared and followed. Specification 6.5.A.4 required surveillance and testing requirements that could have an effect on nuclear safety. Step 45 of Surveillance Test 0379, "Electrical Protection Assembly (EPA) Functional Test," required transfer of RPS bus B to the alternate source using Operations Manual B.9.12-05, Special Procedure G.2.

Contrary to the above, on October 13, 1994, during the performance of the test above, Special Procedure G.2 was not followed in that the B division of reactor building plenum and fuel pool radiation monitors were not bypassed. This resulted in an automatic actuation of an engineered safety feature. This is an example of a violation.

This example is very similar to other examples of failure to follow procedures discussed in Inspection Report No. 50-263/94009. Since the licensee had not completed all of the corrective actions for those previous examples, this event will be considered another example of that violation (263/94009-01e(DRP)).

2.4 Review of Licensee Event Reports (LERs)

2.4.1 (Closed) LER 94-014: Missed Procedural Step While Switching the Power Supply to the Reactor Protection System to the Alternate Source Causes a Partial Containment Isolation This event was discussed in Sections 2.b and 2.c of this report. The event was considered an example of a violation. The corrective actions will be reviewed when the violation is closed. The LER is considered closed to avoid duplication of tracking.

2.4.2 Rotated Fuel Bundle: During refueling operations in the 1994 refueling outage, the licensee discovered fuel bundle LYX927 was rotated 180 degrees from its correct position. At that point, the bundle had not been a part of the refueling operations so it was assumed to have been misorientated during the entire previous cycle. This was verified to be true by a review of the core verification video made at the end of the previous refueling. The misorientation of the fuel bundle was clearly identifiable on that tape. The licensee evaluated the condition and found it not to be reportable. That conclusion was based on the misorientated bundle safety analysis that was performed by NSP Nuclear Analysis and Design on bundle LYX927 for Cycle 16. The analysis demonstrated that compliance with all safety limits was maintained for the entire cycle. The licensee issued Nonconformance Report (NCR) 94-274 to document their assessment and corrective actions. The inspectors reviewed the NCR in detail and determined that the actions taken were acceptable. The licensee's evaluation, disposition, and corrective actions to prevent recurrence were both prompt and thorough. A summary of the licensee's evaluation and corrective actions are presented below:

- At the time of discovery all other fuel bundles in the core were verified to have correct orientation. In addition, the general superintendent operations issued a memo to the operators emphasizing the importance of proper bundle orientation to help preclude another such event.

- The data obtained at the beginning of the last operating cycle for friction testing and scram time testing was reviewed for the cell containing bundle LYX927. There were no discrepancies noted outside normal variances seen between other control blades.

- Bundle LYX927 was inspected via underwater television and no damage was observed.

- Core Reload Verification Procedure 9024 was changed to include a step to specifically verify fuel assembly orientation at the end of the 1994 refueling. The final form this procedure will take was still under review, but the licensee intended to revise the procedure to improve its effectiveness.

- The root cause of the event was human error (inattention to detail). Opportunities to discover the rotated bundle included both the operator and his supervisor when the fuel was originally placed in the fuel cell as well as the three verifiers who performed the Core Reload Verification Procedure. A contributing factor to the event was the repetitive nature of the task.

Technical Specification 6.5 required, in part, that detailed written procedures, including applicable check-off lists and instructions, covering the following, shall be prepared and followed. Specification 6.5.A.2 required fuel handling operations procedures. Procedure 9007, "Procedure for Moving Fuel Into, Out of, and Within the Core," required those activities to be performed in accordance with Operations Manual D.2-05 which specified proper fuel orientation. Failure of the operators to properly implement that instruction was a violation. Core Reload Verification Procedure 9024 was the implementing instruction used to verify proper fuel orientation. Failure of the reviewers to properly implement that instruction was a violation.

These violations were not cited because the licensee's efforts in identifying and correcting the violations met the criteria of the Enforcement Policy, 10 CFR Part 2, Appendix C, section VII.B.

2.5 Plant Startup From Refueling

The inspectors observed major portions of the reactor and plant startup after the 1994 refueling outage. Physics testing and reactor criticality went very smoothly. Communications, supervision, engineering assistance, and monitoring of indications, all of which have been identified as weaknesses in earlier startups, were considered good.

On October 23, 1994, a problem developed just after putting the main generator on line in that arcing was observed on the generator output transformer manual disconnects. The generator had to be taken off line three minutes after being put on line. An investigation found a broken bolt in the linkage for the manual disconnects that had prevented them from fully closing. After that problem was corrected, leakage was noted in the cooling water to the generator exciter causing a further delay. After repairs, the generator was again put on line on October 24 and the plant was brought to full power.

One example of a violation cited in a previous report was identified. Two non-cited violations were identified.

3.0 Maintenance

3.1 Observation of Work

Routinely, station maintenance and surveillance activities were observed and/or reviewed to ascertain that they were conducted in accordance with approved procedures, regulatory guides and industry codes and standards, and in conformance with technical specifications. The following items were considered during this review: approvals were obtained prior to initiating work; test instrumentation was calibrated; functional testing and/or calibrations were performed prior to returning components or systems to service; quality control records were maintained; activities were accomplished by qualified personnel; results were within specification and properly reviewed, and any deficiencies identified were properly resolved. The following maintenance and surveillance activities were observed:

- Test 1227 — ATWS System RPT and ARI Functional Test
- Test 0036-2 — ECCS Auto Initiation Test Including Loss of Auxiliary Power
- WRA 94-05323 — Repair Tack Welds on Adjusting Screws for Jet Pumps
- Procedure 9007 — Procedure for Moving Fuel Into, Out of, and Within the Core
- Mod 93Q180 — Testing of Modified Reactor Pressure/Level Instrument Lines
- 4292PM — Replace Scram Valve Diaphragm
- WRA 94-05366 — Investigate DC Ground on Annunciator Cabinet
- Test 9079 — Benchmark Critical
- 0255-10-1A-4 — Reactor Building to Torus Vacuum Breaker Mechanical Exercise
- WRA 94-05709 — Leak Sealing of High Pressure Turbine Flange

In addition to the jobs listed above, the inspectors observed portions of numerous other refueling outage jobs including core shroud and other in-vessel inspections and reactor reassembly. All work observed was conducted properly with the exceptions discussed below.

3.1.1 Inspection of Core Shroud Welds

The inspectors observed the licensee perform core shroud inspections in response to NRC Generic Letter 94-03, "Intergrannular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors." The inspections were performed by personnel from General Electric (GE) Company. GE personnel used a tracker scanner device to inspect the shroud welds. The tracker scanner device rode on the top of the steam dam and had interchangeable arms of various lengths, which were used to reach the horizontal welds on the outside diameter of the shroud located at different heights on the core shroud. GE experienced difficulties with accessibility of the tracker scanner device due to several causes including variances in weld size, indentations in the steam dam due to a previous modification, and relatively small clearances in the annulus region as compared to other boiling water reactors. However, inspections were completed of all accessible portions of welds H1 through H9 with favorable results. The licensee submitted its inspection results to the NRC for review in a letter dated October 25, 1994. The technical staff in NRR were reviewing the results as of the end of this inspection and intended to issue a safety evaluation report addressing the licensee's inspections and analyses with respect to the requirements of the generic letter.

3.1.2 In-Vessel Visual Inspections

The inspectors observed portions of the of in-vessel visual inspections (IVVI). During the outage-extensive IVVI were performed including the following components: guide rod brackets, steam dryer support brackets, feedwater spargers, core spray spargers and piping, top guide, jet pumps, surveillance sample holders and brackets, access hole covers and shroud and shroud shelf. No problems with the inspections were noted.

3.1.3 Jet Pump Adjusting Screw Tack Weld Failures

General Electric (GE) Service Information Letter (SIL) No.574 was issued October 5, 1993, which discussed the discovery of cracked jet pump set screw tack welds at four GE boiling water reactors. The tack welds in question were safety-related because they assured the set screws did not back out, which could have resulted in structurally compromising the jet pumps. That failure mode could impact the jet pumps' ability to maintain core flooding at 2/3 core height during the design basis loss of coolant accident. GE stated in this SIL that such a failure was not a safety concern because jet pump operability was verified on a daily basis. In the event of a jet pump failure, it would be detected during the operability test. Further, such failure would require the unit to commence shutdown immediately per Technical Specifications.

Based on the information presented in the SIL, the licensee planned and conducted as part of their In Vessel Visual Inspection (IVVI) during the 1994 refueling outage an inspection of the tack welds using a remote underwater camera. The results of that inspection revealed that of the 20 available jet pumps and 80 possible welds, 34 welds had cracked. Each jet pump had two adjusting screws and each screw had two tack welds.

Safety Review Item (SRI) 94-017 was generated discussing the repair of the jet pump adjusting screw tack welds. The repair consisted of adding one tack weld where both tack welds had failed on a particular adjusting screw. The tack welds were repaired using an underwater Gas Tungsten Arc Welding (GTAW) process that fused the metal without adding any filler material. There were a total of 16 welds that met the repair criteria and were repaired by this process. The licensee will reverify weld integrity following operation of the current operating cycle (Cycle 17) to determine the need for additional weld repairs or provide justification for operation through Cycle 18.

The inspectors monitored many portions of the repair activities and reviewed both the work request (WRA 94-05323) and attachments including the procedures used to accomplish the task. Through these activities and interviews with the people directly involved in performing the tack weld repairs, the inspectors noted three examples where procedures were not followed. The licensee issued Nonconformance Reports NCR 94-290 and 94-293 to document the discrepancies.

Technical Specification 6.5 required, in part, that detailed written procedures, including the applicable checkoff lists and instructions, covering areas listed shall be prepared and followed. In particular, Specification 6.5.C.3 required preventive and corrective maintenance of plant equipment and systems that could have an effect on nuclear safety.

The rewelding of the cracked tack welds was performed by GE under the oversight of NSP engineering. Attachment 2 to WRA 94-05322 was GE instruction FDI 0382-51847. Step 4.2 of that instruction required that a functional test of the welding procedure and equipment be performed with the welding fixture submerged in water. This step also required the completion of three acceptable test tack welds in accordance with the qualified welding procedure specification and that these welds be broken by a torque test (25 ft-lb).

During the evening of October 10, 1994, the welding machine that was being used was replaced with a different unit. The new unit used the GTAW process; however, it was a significantly different model which used analog switches to set the critical variables. The functional testing required above had not been performed to qualify the welding equipment prior to using the machine for production welds. This is an example of a violation (263/94011-01a(DRP)).

During the first several rewelds performed, problems were encountered in maintaining the arc after initiation. Consequently, essential welding parameters were being varied in an effort to obtain the required arc duration. As a result of conversations with the individuals performing the welds, it became apparent that welds had been performed with the essential variables set to values outside of their qualified ranges. The inspectors were informed that the "current downslope time" had been changed to a value outside its qualified range and that no requalification had been performed.

Attachment 3 to WRA 94-05322, GE procedure 25A5589 rev. 0, "Underwater Tack Welding," required in step 3.4.1 that all production welds shall be performed in accordance with the qualified weld procedure. Additionally the step required that no welding shall be performed using parameters outside the range of the qualified essential variable without requalification of the procedure. However, welds were performed on October 10, 1994, with an essential variable outside of the qualified range. This was an example of a violation (263/94011-01b(DRP)).

GE procedure 25A5589 also required as part of its quality assurance records, step 4.b, that welding settings, including current, voltage, time, and downslope time as a minimum be recorded and maintained. The procedure that was used was originally written for use with the first weld machine. This machine provided the user with a printout of the essential parameters following a weld. However, when the switch was made to the second machine, this data was not recorded as required by the procedure for seven welds performed the evening of October 10, 1994. Consequently, it was not available for review by the inspectors the following morning. This was an example of a violation (263/94011-01c(DRP)).

3.1.4 Work Started on the Wrong Valve

On September 28, 1994, while performing Work Request Authorization (WRA) 94-04755 to disassemble reactor core isolation cooling system air operated check valve AO-13-22, workers mistakenly began to disassemble MO-2565, a motor operated valve on the main steam drain line. Shortly into the job, the workers began to question whether they were on the correct valve and identified their error. They immediately stopped the work and informed their supervisor. The licensee issued Nonconformance Report (NCR) 94-243 to document the investigation and corrective actions. The investigation determined the following:

- Two maintenance workers were working on the job, a junior plant worker and a more senior traveling worker. An unclear division of authority between the workers regarding who was responsible for identifying the correct valve may have occurred.

- Neither worker had performed an adequate review of the work package prior to beginning the work. Thus they didn't realize that they were to work on an air operated valve not a motor operated one.

- Neither worker performed adequate self-checking during the work to verify that the valve they located was the one listed in the work package.

- The workers had to stop the work twice before they actually got to the point of starting the disassembly, once to change the radiation work request because it listed the wrong room, and once when the lead worker was called away for a fitness for duty test just as they located the valve.

The licensee had taken the following immediate corrective actions:

- Secondary containment integrity was verified. It was determined that opening MO-2565 did not violate secondary containment integrity.

- All mechanical maintenance work was stopped until group meetings were held with both shifts of maintenance workers to discuss the event. Results of the discussions were issued as a "Maintenance Notes" document for all maintenance workers.

- The two workers were disciplined.

- Other training and procedural corrective actions were being developed.

Technical Specification 6.5 required, in part, that detailed written procedures, including applicable check-off lists and instructions, covering the following, shall be prepared and followed. Specification 6.5.C.3 required preventive or corrective maintenance of plant equipment that could have an effect on nuclear safety. Work Request Authorization 94-04755 required disassembly for an internal inspection of air operated check valve AO-13-22.

Contrary to the above, on September 28, 1994, during the performance of the work above, the WRA was not followed in that motor operated valve MO-2565 was partially disassembled. This is an example of a violation.

This example is very similar to other examples of failure to follow procedures discussed in Inspection Report 263/94009. Since the licensee had not completed all of the corrective actions for those previous examples, this event will be considered another example of that violation (263/94009-01f(DRP)).

3.1.5 <u>Inadequate Surveillance Testing Procedures</u>

As discussed in Section 2.2.4 of this report, on October 17, 1994, the licensee experienced an unexpected isolation of the shutdown cooling suction valves on a high pressure signal. The event was due to an inadequate procedure for backflushing the reactor pressure and level instrument lines as part of Test 0255-20-1D-1, "Excess Flow Check Valve Test."

Performance of the procedure could have been expected to cause the isolation of shutdown cooling but the procedure had no provisions for preventing it. The procedure did require that the residual heat removal pump (and other emergency core cooling system pump) control switches be in the "Pull-to-Lock" position to prevent actuation on low reactor level. However, closing of the suction valves is initiated from a separate high pressure signal. Closing of the suction valves had little safety significance since the procedure required the pumps to be out of service but it was an unnecessary challenge to part of a safety system.

The licensee issued LER 94-016 concerning this event in which they committed to revise the test procedure to ensure the shutdown cooling isolation valves are closed during testing that could cause automatic actuation.

As discussed in Section 2.2.5 of this report, on October 23, 1994, the licensee experienced an unexpected isolation of the steam supply valves for the high pressure coolant injection (HPCI) turbine during the performance of Test 0255-06-IA-1, "HPCI System Tests with Reactor Pressure at Rated Conditions." After extensive investigations and calculations the licensee determined that the isolation could have been expected under the circumstances of the test. The test, conducted at reduced reactor pressure (about 900 psig) during the plant startup, went beyond the technical specification operability requirements and attempted to establish full reactor pressure (about 1000 psig) HPCI pump discharge and turbine RPM conditions in order to meet ASME Section XI testing requirements. This resulted in abnormally high steam flow to the turbine. It was later determined that the turbine RPM indication was also somewhat out of calibration which contributed to the problem.

Later testing confirmed that the HPCI system was not damaged by the high steam flow. The root cause of the event was an inadequate procedure for the plant conditions. The licensee intended to issue LER 94-017 as a written followup. The inspectors discussed the anticipated corrective actions with the licensee. Corrective actions were expected to consist of procedure revisions and a possible amendment request for Technical Specification testing requirements.

10 CFR 50, Appendix B, Criterion V, required, in part, that activities affecting quality shall be prescribed by documented instruction, procedures, or drawings, of a type appropriate to the circumstances.

Contrary to the above, on October 17, 1994, the licensee performed Test 0255-20-1D-1, an activity affecting quality, with a procedure that was not appropriate to the circumstances in that it did not prevent the inadvertent isolation of shutdown cooling suction valves.

Contrary to the above, on October 23, 1994, the licensee performed Test 0255-06-1A-1, an activity affecting quality, with a procedure that was not appropriate to the circumstances in that it did not prevent an inadvertent isolation of HPCI steam supply valves.

These violations were not cited because the licensee's efforts in identifying and correcting the violations met the criteria of the Enforcement Policy, 10 CFR Part 2, Appendix C, Section VII.B.

3.1.6 Completion of Refueling Outage

During this inspection period the licensee completed a scheduled refueling outage on the original 39-day schedule. The licensee completed 1,244 work request authorizations, 34 modifications, and a large number of preventive maintenance and surveillance tasks. The work was completed on schedule despite major delays in core shroud inspection activities and jet pump tack weld repairs. Along with almost all of the originally scheduled work, the licensee completed 225 emergent work requests including repairs to the inboard main steam isolation valves, the main generator exciter rotor windings, and the above mentioned jet pump tack welds.

Communications, teamwork, scheduling, risk control, and industrial safety were all considered strengths of the outage performance. No significant industrial safety events or equipment damage occurred. Several innovations were implemented to more efficiently complete critical path work such as reactor disassembly and reassembly and equipment lineup checklists.

The involvement of the quality services group in self-assessment of maintenance, modification, and testing activities during the outage was also considered strong. The inspectors and auditors activities appeared to be both broader and more in depth than in previous outages. As discussed in Inspection Report No. 50-263/94009, Section 6, the findings of the quality services group were distributed and discussed in a more timely manner during the outage.

3.1.7 <u>(Closed) Temporary Instruction (TI) 2515/125 - Foreign Material Control</u>: The NRC issued this TI on August 25, 1994, to provide guidance for determining whether licensees had implemented effective procedures to prevent foreign material from inadvertently entering safety systems during maintenance activities, outages, and routine operations. Inspection requirements and results were as follows:

<u>Procedure Review</u>: The inspectors determined that the licensee's work control procedures and practices adequately addressed foreign material control. The basic requirements were contained in Administrative Work Instructions 4 AWI-04.05.09, "Foreign Material Exclusion/Cleanliness Control," and 4 AWI-04.02.01, "Housekeeping." Other more specific requirements were located in other appropriate documents such as those for reactor disassembly and reassembly, fuel movement activities, and significant maintenance activities. Cleanliness control and inspection requirements were generally included in the work instructions for work request authorizations.

In addition to procedures covering maintenance work, the licensee developed specific requirements in response to emergency core cooling system (ECCS) strainer plugging concerns discussed in NRC Bulletin 93-01, Supplement 1. Requirements included adding ECCS strainer integrity and cleanliness inspections to procedure 1132, "Pressure Suppression Chamber Internal Structural Visual Inspection," adding a QC witness point to verify all foreign material has been removed from the torus in the torus manway section of procedure 8080, "Primary Containment Hatch Closure Procedure," and developing a new procedure 1371, "Drywell Prestart Inspection." This last procedure contained steps to document a complete inspection of the drywell for an loose material.

<u>Applicability</u>: The inspectors determined that the licensee's procedures were applicable to the various types of work activities for which foreign material control was appropriate.

<u>Experience</u>: Neither the inspectors nor licensee maintenance personnel interviewed were aware of any incidents caused by foreign material during the last year which resulted in equipment damage or operability concerns. Inspection Report No. 50-263/94007, Section 2.d, discussed concerns with cleanliness control in the area around the spent fuel pool during new fuel inspections. The report also discussed licensee corrective actions. The inspectors noted increased vigilance on the

part of licensee employees to those concerns during the refueling outage.

Observation: The inspectors observed activities during the 1994 refueling outage and determined that foreign material exclusion control procedures were generally being followed. The inspectors toured the inside of the torus suppression chamber both when water was pumped down for inspection of the ECCS suction strainers and just before final closeout. No debris was observed in the torus and cleanliness of the suction strainers was excellent.

The inspectors also toured the drywell several times during the outage including shortly before final closeout. The licensee performed a modification during the outage to remove all remaining fibrous insulation from systems in containment and replace it with insulation of a type that would not interfere with the ECCS suction strainers. Minor concerns regarding other material such as small pieces of duct tape left attached to various pipes and supports were brought to the attention of the licensee by the inspectors and rapidly corrected. The licensee had not yet completed its final inspection at that time. Overall cleanliness of the containment was good with no temporary equipment left inside after closeout.

The licensee experienced a few instances of foreign material dropping into the reactor cavity or spent fuel pool during the outage. The events appeared not to be related to improper foreign material control but rather happened despite reasonable efforts. The licensee recovered the items except for one case where a small retaining clip from the reactor head O-ring became detached, probably during head removal. This event was discussed in Inspection Report No. 50-263/94009, Section 3.c. The clip was never found. The licensee performed an evaluation and determined that even if it was in the reactor it would not be a safety concern.

This temporary inspection is considered closed.

3.1.8 Review of Licensee Event Reports (LERs)

(Closed) LER 94-009: Electrical Maintenance Personnel Error During Surveillance Test Causes Both Emergency Diesel Generators to Fast Start: This event was discussed in Inspection Report No. 50-263/94009, Section 3.d. It was considered one example of a violation (263/94009-01c). Completion of corrective actions for the event will be reviewed when the violation is closed. The LER is considered closed to avoid duplication of tracking.

(Closed) LER 94-015: Failed Component in the Uninterruptible Power Supply Causes a Partial Containment Isolation: This event was discussed in Section 2.2.3 of this report. The cause of the event appeared to be a random component failure of an electrical static switch logic card. The licensee's corrective actions were adequate. This event is considered closed.

Three examples of a cited violation were identified. An additional example of a violation cited in a previous report was also identified. Two non-cited violations were identified.

4.0 Engineering

4.1 Review of Licensee Event Reports (LERs)

4.1.1 <u>(Closed) LER 93-001: Potential Single Failure of Standby Gas Treatment Room Heater Could Cause Temperatures Above Equipment Ratings for Both Standby Gas Treatment Trains</u>: This issue was previously discussed in Inspection Report No. 50-263/92019, Section 2.b.(2). During the 1994 refueling outage Modification 93Q325 was completed to take the place of a temporary bypass that had been in place to resolve the issue. The modification completed the licensees corrective actions for the LER and this issue is considered closed.

4.1.2 <u>(Closed) LER 93-008: Reactor Protection System Actuation From Low Reactor Water Level Caused by Condensate Pump Trip</u>: This event was previously discussed in Inspection Report No. 50-263/93014, Section 2.b.(5). At the time of the event no definite cause could be found for the trip. During the 1994 refueling outage the licensee performed preventive maintenance and inspection of the motor breaker and found no problems. The motor was replaced with a spare during the outage and the licensee planned to inspect it for possible internal damage. The inspectors determined that all reasonable efforts had been taken to try to find the cause of the breaker trip. No additional problems occurred with the breaker in over one year of operations since the event. This issue is considered closed.

4.1.3 <u>(Closed) LER 94-008 and (Closed) LER 94-008, Revision 1: Structural Beam Connections Associated With the Cable Spreading Room Floor Found to be Different Than Design</u>: This issue was previously discussed in Inspection Report No. 50-263/94007, Section 4.a, and was also being tracked as Unresolved Item 263/94007-02(DRS)). The issue was still being reviewed by the NRC as discussed below. The original LER and revision are considered closed to avoid duplicate tracking.

4.1.4 <u>(Closed) LER 94-010: Containment Isolation Valves Exceed Local Leak Rate Limits</u>: This event was previously discussed in Inspection Report No. 50-263/94009, Section 4.a. During the 1994 refueling outage modification were accomplished to replace the outboard main steam isolation valves with a different type of valve with better isolation characteristics and add a safety grade air supply to the inboard main steam isolation valves which should supply a long term post accident air source to help seal the inboard valves. Thus the licensee can take credit for air availability and use it during future local leak rate tests. The three inboard valves which failed their leak rate tests were also repaired including installing a new seat on the "A" valve. All valves passed an "as left" local leak rate test. The modifications and repairs should greatly improve the reliability of the valves.

A separate issue discussed in this LER related to the improper local leak rate testing of the "B" Feedwater (FW) line check valve, FW-94-2. As discussed in the LER the engineer who was using procedure 0137-08-2, "B Loop Feedwater Check Valves FW-94-2 and FW-97-2" did not take the procedure into the area where the valves were located. This test was meant to demonstrate primary containment integrity with respect to the feedwater check valves in the "B" line. However, the engineer manipulated the wrong valves and the test failed.

Technical Specification 6.5 required, in part, that detailed written procedures, including applicable check-off lists and instructions, covering the following, shall be prepared and followed. Specification 6.5.A.4 required surveillance and testing requirements that could have an effect on nuclear safety.

Contrary to the above on October 7, 1994, steps 7 & 8 of procedure 0137-08-02 to line up the system for a local leak rate test on the "B" feedwater line were not properly followed. This is a violation. However, this event was self-identified and of low safety significance since later disassembly of the valve showed that no problems existed with FW 94-2 and additionally the redundant check valve in the "B" feedwater line did pass its local leak test. The licensee's corrective actions, as discussed in the LER, were adequate and have been completed. Therefore, this violation will not be cited in accordance with Section VII.B of the Enforcement Policy. This issue is considered closed.

4.1.5 <u>(Open) LER 94-011: Inoperable Safety/Relief Valves Resulting in Violation of Plant Technical Specifications</u>: This LER described licensee findings that two of the eight safety/relief valves on the main steam lines were potentially inoperable for self-actuation at the same time. One valve had displayed a bellows leak alarm for many months during the last operating cycle. It was considered inoperable but Technical Specifications allowed continued operation with seven of the eight valves operable. During the 1994 refueling outage, bench testing of another valve determined that its self-actuation set point was slightly higher than the 1% tolerance allowed over the normal setpoint. Thus Technical Specifications were potentially not met during the operating cycle.

The LER discussed licensee analysis which demonstrated that all safety limits would be met even with three safety valves totally inoperable for self-actuation. In this case both of the valves would still have worked but would have lifted at slightly over the normal setpoints. Thus this finding had a low safety significance. The problem was identified by the licensee as part of required testing. Corrective actions included replacing the topworks for both of the valves. The licensee intended to disassemble and inspect both of the failed topworks to determine the root cause of the failures. That work was scheduled for the first quarter of 1995. The licensee intended to update the LER with the results of that work. This LER is considered open pending the inspectors review of the root cause of the problems with the safety valves.

4.1.6 (Closed) LER 94-012: Fire Door Latch Found with Insufficient Throw Due to Original Construction Error: On September 30, 1994, the licensee discovered that a fire door did not have the required door latch throw. The door latch present was 5/8 inch and the required latch should have been 3/4 inch in size. This determination was made during the Triennial Fire Protection Inspection required by Appendix C of the Operational Quality Assurance Plan. Technical Specification 3.13.G.1 required all penetration fire barriers in fire boundaries to be operable whenever safe shutdown equipment in that fire area was required to be operable. At the time of discovery, the plant was shut down; however, it was believed this deficiency existed since original plant construction.

Corrective actions to restore the barrier were immediately initiated and have been completed. In addition, all fire doors used in fire area barriers were inspected to ensure that all latch throws were in compliance with the requirements of NFPA 80. This violation was licensee-identified and of relatively low safety significance due to a low volume of combustibles in the area, and a continuous fire detection monitoring system being used in the affected zones. Thus, a Notice of Violation will not be issued in accordance with the criteria of the Enforcement Policy, 10 CFR Part 2, Appendix C, section VII.B.

4.2 Followup of Previous Inspection Issues

4.2.1 (Closed) Inspection Followup Item (263/92013-02(DRP)): Testing of Emergency Diesel Generators at Less Than Design Power Factor (pf): This issue was previously discussed in Inspection Report No. 50-263/92013, Section 3.a.(2). The licensee provided the emergency core cooling system loads for the #12 emergency diesel generator (EDG) as the most limiting case. The calculation covered the injection and recirculation phases of the worst case accident operating scenario. The following #12 EDG loads were identified:

Operating Phase	KW	KVA	Overall pf
Injection	2211.5	2406	0.92
Recirculation	2193.5	2400	0.91

Other small loads may be restored at the operator's discretion up to the EDG's 2500 KW continuous rating. In addition, the EDGs were qualified to operate for 2000 hours at a 10% overload (2750 KW).

The inspectors reviewed monthly surveillance test 0187-2, "12 Emergency Diesel Generator/12 Emergency Service Water Pump System Tests," Revision 17. The #12 EDG was loaded to between 2400 and 2600 KW for 1 hour with the pf adjusted to approximately 1.0. The procedure also verified that 2500 KW had been obtained. Since 2500 KW is equal to 2500 KVA at a 1.0 pf, the test bounded the calculated total accident KVA loads. This item is considered closed.

4.2.2 (Open) Unresolved Item (263/94007-02(DRS)): Structural Beam Connections Associated With the Cable Spreading Room Floor Found to be Different Than Design: This issue was previously discussed in Inspection Report No. 50-263/94007, Section 4.a, and Licensee Event Report 94-008, Revision 1. During this inspection period an NRC specialist inspector reviewed the structural calculations associated with the long term resolution of this issue and provided the following comments to the licensee:

- The use of higher allowable stresses for a Halon system actuation was not well established. The assumption that this was an infrequent emergency event was questionable based on a recent inadvertent Halon discharge in the plant's process computer room. Pending additional review by the NRC, this aspect will remain unresolved.

- Seismic and Halon system discharge loads were not considered concurrently, and the basis for this was not established. Since the Halon system instrumentation was not specifically qualified for seismic loading, the potential exists for this interaction. Pending additional review by the licensee, this aspect will remain unresolved.

- The assertion that Monticello's licensing basis did not require the tornado differential pressure to be applied to the cable spreading room floor appears to be inconsistent with the Monticello Final Safety Analysis Report. Pending further clarification by the licensee, this aspect will remain unresolved.

- The use of "aged concrete strength" instead of the 28-day strength for the long term resolution of this problem was a departure from the criteria given in Monticello's Updated Safety Analysis Report. Pending further review by the NRC for the acceptability of the licensee's methodology, this aspect will remain unresolved.

4.3 Plant Improvement Modifications

The license completed 34 modifications during the 1994 refueling outage. Several of the modifications were implemented to improve reliability of important safety equipment based on the licensee's individual plant examination, design basis review, and recent industry events. Among those types of modifications were the following:

- Replacement of all remaining fibrous insulation in the drywell.
- Enhancements to several motor operated valves.
- Replacement of outboard main steam isolation valves with models less susceptible to leakage.
- Addition of a safety grade air supply to the inboard main steam isolation valves to improve long term post accident leak tightness.

- Addition of a safety grade air supply to all safety relief valves for long term post-accident operability.
- Several enhancements to the control room emergency filtration train system to improve the reliability of the post accident control room habitability system.
- Addition of loss of power indicating lights to control room annunciator panels.
- Replacement of all scram pilot solenoid valves with valves containing improved diaphragms.
- Piping modification to reduce vibrations and improve reliability of control rod drive pumps.

The modifications demonstrated a strong commitment on the part of the licensee to continuous improvements in safety.

4.4 Licensee Review of Industry Events and Issues

In the daily interactions with licensee personnel the inspectors noted a strong and timely program for licensee review of industry events and issues. From time to time the inspectors discussed emerging issues brought to their attention through NRC morning reports or preliminary notifications with the appropriate system engineers. Invariably the system engineers had already heard about the issues and were reviewing them for applicability to the site. The licensee's use of electronic mail to rapidly distribute information regarding industry events appeared to be working well.

Two non-cited violations were identified.

5.0 Plant Support

5.1 Radiation Protection

5.1.1 Clean Area Maintained on Refueling Floor

The inspectors noted during observations of the core shroud inspections and in-vessel visual inspections that a portion of the refueling floor was maintained as a clean area in which access was permitted in street clothing. This area was used for operation of recording and other electronic equipment, as well as supervision and observations of the activities. Although it was an extra effort to maintain the area decontaminated during in-vessel work, it eased several aspects of the job. It also reduced the amount of anti-contamination clothing needed and reduced radioactive waste generated.

5.1.2 Self Assessment of September 1994 Snubber Replacement

The inspectors reviewed the licensee's self assessment of a drywell snubber replacement which occurred on September 19, 1994, in which two workers involved in the evolution exceeded their electronic dosimeters' 150 millirem (mrem) (1.50 millisievert (mSv)) accumulated dose alarms by

23 and 108 mrem (0.23 and 1.08 mSv). The preliminary investigation appeared to be thorough and identified weaknesses on the part of the maintenance workers and the radiation protection specialists (RPS).

The licensee's evaluation indicated that the workers were not fully cognizant of the alarm threshold of their electronic dosimeters (EDs) and failed to exit the area when the accumulated dose alarm sounded. Prior to their EDs alarming, an RPS observed that the workers were not knowledgeable of their allowed dose. After verifying the workers' dose limit, the RPS had the workers exit the drywell. During the time that the RPS obtained the workers dose information, the workers exceeded the dose alarm threshold but remained at the work location, failing to respond properly to the dose alarms. The licensee's investigation indicated that the workers were confused by the difference between the dose alarm, which required immediate exit of the area, and the dose rate alarm, which required moving away from the source. Additionally, the audibility of the alarm was lessened by the background noise level. Following the event, the licensee provided additional training to personnel on ED alarms and appropriate personnel responses. The licensee also planned to investigate ED alarms for more distinct, audible alarm tones.

Procedure 4 AWI-08.04.02, "Personnel Exposure Monitoring and Control," Revision 1, required workers to immediately exit an area when their dose alarms sound and the displayed doses exceed the alarm thresholds. Procedure 4 AWI-08.04.02 implemented the external radiation monitoring requirements of the Radiation Protection Plan as required by Technical Specification (TS) 6.5.B. The failure of the maintenance workers to immediately exit the area upon receiving dose alarms would be a violation of TS 6.5.B. However, since the violation was identified by the licensee and corrective actions were implemented, the violation meets the criteria contained in Section VII.B. of 10 CFR Part 2, Appendix C and is not cited.

The inspectors discussed additional aspects of the snubber job evolution with one of the maintenance workers, who indicated that there were communications problems between the RPSs and the two maintenance personnel, which contributed to some confusion concerning the removal of lead shielding by the maintenance workers. Also, the RPS at the drywell did not fully understand the workers intent to enter a new work location, which required additional radiological surveys. The inspectors discussed this event with plant management. Although the event was isolated and the investigation appeared to be thorough, the personnel actions demonstrated an overall lack of good radiation protection knowledge by the maintenance workers.

5.1.3 <u>Evaluation of Americium-241 Contamination During Main Steam Isolation Valve Replacements</u>

The inspectors reviewed the licensee's planning and execution of the replacement of inboard and outboard main steam isolation valves (MSIVs) and identified two examples of inadequate surveys of radiological hazards performed by the licensee.

- During the removal of the outboard MSIVs, the licensee detected elevated concentrations of americium-241 (^{241}Am) in the oxidation layer of the main steam piping. Although initial surface contamination surveys performed on September 25, 1994, indicated an alpha contamination component, the radiation protection (RP) staff did not investigate the contamination as required by plant procedures. Consequently, the RP staff did not assess the extent of alpha contamination in the evaluation of radiological requirements, including the decision not to require respiratory protection. During the evolution, the licensee provided good RP coverage and air sampling. The air sampling did not detect alpha contamination during work on MSIVs A, B, and C; however, during the grinding of the inner diameter of the D main steam line on September 29, 1994, an air sample indicated an airborne radioactivity concentration of about 304 derived air concentration (DACs) total gamma radioactivity, of which 303 DACs were ^{241}Am. The grinding was stopped, and workers were evacuated. For subsequent MSIV work, the licensee implemented additional engineering controls and more conservative respiratory protection requirements. Although no significant exposures resulted from the grinding evolution (Section 5.1.5), the licensee failed to adequately assess the alpha radioactivity component in the pre-job planning.

- Following the outboard MSIV evolution, work was performed on the inboard MSIVs using airborne isolation boundaries, high efficiency particulate air (HEPA) filtration, and respiratory protection. However, respiratory protection (i.e. a particulate air respirator) was required only during grinding evolutions and as per posted instructions. Dust masks were required for all additional entries. Following the grinding performed on October 2, 1994, two persons entered the MSIV area for inspections with dust masks, as allowed by the radiation work permit (RWP), and were in the area for less than five minutes. An air sample collected during the grinding operation indicated an elevated airborne contamination level of about 2500 DACs. Following the workers' exit, an additional air sample taken resulted in less than detectable airborne radioactivity. However, the licensee failed to provide an adequate survey prior to allowing personnel into the area without respiratory protection. The licensee immediately revised the RWP to require air samples prior to entrances without respirators.

10 CFR 20.1501(a) requires that surveys be made to comply with regulations in Part 20 and to evaluate the extent of radiation levels, quantities of radioactive material, and the potential radiological hazards present. Specifically, surveys were required to comply with the occupational dose limits contained in 10 CFR 20.1201. The failure to

perform adequate surveys during the respiratory protection evaluations for the outboard and inboard MSIV grinding work to ensure that occupational dose limits were not exceeded is a violation of 10 CFR 20.1501(a) (263/94011-02(DRSS)).

Although these examples were licensee identified events and the licensee implemented immediate corrective actions, the violation is cited because the first example, concerning the planning of the outboard MSIV evolutions, indicated a lack of attention to procedural requirements similar to a previous violation (Violation No. 50-263/94006-01(DRSS)) and because the second example, concerning the inadequate inboard MSIV RWP requirements, should have been prevented by the licensee's immediate corrective actions for the first example.

The licensee was investigating the source of the ^{241}Am in the corrosion film. Although the licensee had not had fuel performance problems in several years, the most probable source of the ^{241}Am was plate-out of activation/fission products from historical fuel problems. The licensee had enlisted a consultant to provide an additional, independent review of the data. The results of the licensee's investigation will be reviewed as a routine part of future inspections.

5.1.4 Control of Airborne Radioactive Materials And Respiratory Protection

The inspectors reviewed weaknesses in the licensee's use of engineering controls and respiratory protection during the MSIV replacement evolution. The licensee did not provide adequate engineering controls, and a respirator failed during grinding of the inboard MSIV, which appeared to result from weaknesses in the licensee's procedures.

The licensee established HEPA filtration units for the inboard and outboard MSIV replacements to limit the possibility of airborne contamination. The work consisted of heavy brushing and grinding of the internals of the piping, creating a high concentration of airborne particulates. However, the licensee failed to provide adequate surveillance of the systems to ensure that the filters remained operable. During the evolution, the HEPA filtration unit located in the drywell became loaded with particulate material, which resulted in decreased flow in the system. An RPS noticed the decreased flow and alerted the RP staff of the problem. However, grinding of the contaminated MSIV had already taken place during the low flow condition, resulting in a very high airborne radioactivity concentration while personnel were present. The use of respiratory protection limited the impact of the airborne radioactivity contamination.

The licensee implemented adequate, immediate corrective actions. For subsequent work, the applicable RWP contained the requirement for a continuous monitor at the HEPA unit and an immediate work stoppage if the unit lost flow or failed.

Engineering controls to limit airborne radioactivity are required by 10 CFR 20.1701. The failure of the licensee to provide adequate engineering controls would be a violation of 10 CFR 20.1701. However, since the violation was identified by the licensee and corrective actions were implemented, the violation meets the criteria contained in Section VII.B. and is not cited.

During grinding of an inboard MSIV, a connection on a maintenance worker's supplied air respirator (SAR) became uncoupled, when a clamp holding the air hose to a fitting came loose in the assembly. The licensee's evaluation indicated that the connection was not the proper one for the apparatus, in that the hose/fitting assembly connecting the face mask to the air regulator was not designed for a SAR. Due to inadequate procedural guidance, the line designed for a powered air purifying respirators (PAPR) had been installed. Both lines were supplied and individually approved by the vendor, and, according to the vendor, the use of the PAPR hose/coupling assembly should not have affected respirator performance. However, the incorrect assembly from the mask to the regulator was not in accordance with the National Institute for Occupational Safety and Health/Mine Safety and Health Administration (NIOSH/MSHA) approval for the SAR.

10 CFR 20.1703(a)(1) requires that respiratory equipment used by a licensee have certification by the NIOSH/MSHA. The licensee planned to revise the applicable procedure and to provide a demonstration model/display to ensure that the proper respirator assembly occurs in the future.

The failure of the licensee to use respiratory equipment which has NIOSH/MSHA certification would be a violation of 10 CFR 20.1703(a)(1). However, since the violation was identified by the licensee and corrective actions were implemented, the violation meets the criteria contained in Section VII.B. and is not cited.

5.1.5 Internal Dose Assessment of Potential Americium-241 Intakes

The licensee performed excellent assessments of potential internal exposures. Following the potential internal exposures from the MSIV replacement work, the licensee performed onsite whole body counting (WBC) for all personnel involved in the evolutions and sent more potentially contaminated personnel offsite for more sensitive WBC for ^{241}Am. The WBC results did not indicate any measurable intakes of cobalt-60 (^{60}Co) or ^{241}Am, but select personnel had small positive results for zinc-65 (^{65}Zn) and manganese-54 (^{54}Mn). The licensee provided additional WBC to verify the positive results. Although the licensee did not formally have a program developed, in vitro bioassay was also implemented to achieve lower minimum detectable activities (MDAs) for ^{241}Am than WBC.

The inspectors reviewed the preliminary results of the in vitro bioassay measurements for the workers involved. The highest measured dose from ^{241}Am was about 30 $\pm$ 3 mrem (0.30 $\pm$ 0.03 mSv) committed effective dose equivalent (CEDE). The licensee continued to obtain and evaluate data.

The final dose calculations and estimates will be reviewed during future routine radiation protection inspections.

5.1.6 External Dose Control and As-Low-As-Reasonably-Achievable (ALARA)

Dose control for the outage was very good. The accumulated dose for the refueling outage was 330 rem (3.3 Sv) versus a pre-outage goal of about 365 rem (3.65 Sv). The licensee's 1994 annual dose goal was about 465 rem (4.65 Sv), which appeared to be attainable based on the current accumulated dose.

The inspectors discussed the licensee's ALARA policies with the superintendents of the operations, maintenance, and engineering departments. Each department manager had a very good understanding of the plant's ALARA principles and was conscious of the department's dose and dose tracking. Interviews with plant workers indicated a fair understanding of RWP requirements and radiation work practices. Additionally, the licensee began a program to evaluate the radiation source term, which included a inventory of valves which contribute to ^{60}Co activity in the reactor coolant system.

5.1.7 Radiological Confirmatory Measurements and Chemistry Quality Control

Five samples (reactor crud filter, particulate air filter, charcoal filter cartridge, primary coolant, and offgas) were analyzed by the licensee and in the Region III mobile laboratory for gamma emitting radionuclides. The air filter sample was one of the licensee's air samples described in Section 5.1.3 of this report. Comparisons were made on a random selection of the licensee's three high purity germanium detectors. Additionally, an air particulate filter standard was counted on each of the licensee's detectors. In 113 comparisons (Table 1), all of the licensee's analyses were in agreement with the NRC results.

A simulated liquid waste sample will be analyzed by the licensee for gross beta, iron-55, strontium-89, strontium-90, and hydrogen-3 activity. A portion of this sample will be analyzed by the NRC reference laboratory, and the results will be compared during a future confirmatory measurements inspection.

The inspectors observed licensee personnel collecting and preparing samples. Overall, the chemistry technicians demonstrated good technique and radiation protection practices. The licensee's preparation, maintenance, and review of performance trend charts and performance in the interlaboratory comparison program were very good. The licensee's quality control of the post accident sampling system (PASS) was also very good. Isotopic comparisons between samples obtained at PASS and routine sampling points indicated that the PASS was representative of the bulk reactor coolant.

5.1.8 Radiological Environmental Monitoring Program (REMP)

The inspectors reviewed the 1993 Annual Radiological Environmental Operating Reports. The report contained sample collection and analysis results as required by the licensee's TS and Offsite Dose Calculation Manual (ODCM). All samples were below TS reporting limits. Although an air sampling station lost power during a week of operation, a thermoluminescent dosimeter was the only sample listed as a missed sample in the report. As only a small quantity of air was passed through the sample media, a representative sample was not obtained for the sample period. The sampling anomaly was noted in the data tables, but a comment in the missed sample listing was not made. The licensee acknowledged the inspectors' comment and agreed to include this type of sample anomaly in future reports.

The inspectors observed an RPS during the routine collection of air and water samples. Air samplers were operable, in good condition, and within calibration. The flow meter used in the sample collection was also in calibration. The RPS demonstrated good technique and verified that no air inleakage was present after installing new air filter cartridges. However, the sample collection device for river water sampling was degraded. The licensee acknowledged the inspectors concerns and planned to replace the collection device to decrease the likelihood of sample contamination.

5.1.9 Followup of Previous Inspection Items

(Closed) Inspection Follow-up Item No. 50-263/93011-01: Nonradiological Split Samples: The licensee was to analyze a split sample for chloride, sulfate, and fluoride and report the results to the Region III office for comparison. The licensee's results appeared to be consistent with the prepared concentrations. Due to the poor NRC analysis statistics, the NRC reference laboratory's results could not be compared. This item is considered closed.

One cited violation, with two examples, was identified. In addition, three non-cited violations were identified.

6. Non-cited Violations

The NRC uses the Notice of Violation as a standard method for formalizing the existence of a violation of a legally binding requirement. However, because the NRC wants to encourage and support licensee initiatives for self-identification and correction of problems, the NRC will not generally issue a Notice of Violation for a violation that meets the tests of 10 CFR 2, Appendix C, Section VII.B. These tests are:

a. it was not a violation that could have reasonably been prevented by corrective action to a previous violation;

b. the violation was not of major safety significance;

c. the violation was or will be corrected, including measures to prevent recurrence, within a reasonable time; and

d. it was not a willful violation.

Violations of regulatory requirements identified during this inspection for which a Notice of Violation will not be issued are discussed in Sections 2.4.2, 3.1.5, 4.1, 5.1.2 and 5.1.4 of this report.

7. <u>Management Changes</u>

On November 9, 1994, the licensee announced that Lon Waldinger, General Manager Monticello Site, had been selected as Director of Generation, New Business Development. After a transition period the General Manager Monticello Site position was to be eliminated and the responsibilities assumed by the Plant Manager.

8. <u>Exit Interview</u>

The inspectors met with the licensee representatives denoted in paragraph 1 at the conclusion of the inspection on November 15, 1994. The inspectors summarized the purpose and scope of the inspection and the findings. The licensee strengths and weaknesses identified in the report were discussed. The inspectors also discussed the likely informational content of the inspection report, with regard to documents or processes reviewed by the inspectors during the inspection. The licensee did not identify any such documents or processes as proprietary.

Attachments:
Table 1 - USNRC Sample Comparison Results
Attachment 1 - Criteria for Comparing Analytical Measurements

TABLE 1

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

FACILITY: MONTICELLO

FOR THE 3RD QUARTER OF 1994

SAMPLE	NUCLIDE	NRC VAL.[1]	NRC ERR.[1]	LIC.VAL.[1]	RATIO[2]	RES[3]	RESULT[4]
AIR FILTER STAND. DET.#1	CO-57	1.37E-02	4.39E-04	1.37E-02	1.00	31.2	A
	CO-60	3.16E-02	6.91E-04	3.28E-02	1.04	45.7	A
	SR-85	5.03E-02	1.62E-03	5.17E-02	1.03	31.0	A
	Y-88	7.06E-02	1.84E-03	7.62E-02	1.08	38.4	A
	CD-109	6.60E-01	3.82E-02	6.75E-01	1.02	17.3	A
	SN-113	4.07E-02	1.59E-03	4.15E-02	1.02	25.6	A
	CS-137	1.98E-02	6.07E-04	1.98E-02	1.00	32.7	A
	CE-137	2.15E-02	3.71E-03	2.16E-02	1.01	5.8	A
	HG-203	4.66E-02	2.16E-03	4.68E-02	1.00	21.6	A
AIR FILTER STAND. DET.#2	CO-57	1.37E-02	4.39E-04	1.34E-02	0.98	31.2	A
	CO-60	3.16E-02	6.91E-04	3.07E-02	0.97	45.7	A
	SR-85	5.03E-02	1.62E-03	4.86E-02	0.97	31.0	A
	Y-88	7.06E-02	1.84E-03	7.29E-02	1.03	38.4	A
	CD-109	6.60E-01	3.82E-02	6.67E-01	1.01	17.3	A
	SN-113	4.07E-02	1.59E-03	4.07E-02	1.00	25.6	A
	CS-137	1.98E-02	6.07E-04	1.89E-02	0.95	32.7	A
	CE-137	2.15E-02	3.71E-03	2.13E-02	0.99	5.8	A
	HG-203	4.66E-02	2.16E-03	4.38E-02	0.94	21.6	A
AIR FILTER STAND. DET.#3	CO-57	1.37E-02	4.39E-04	1.47E-02	1.08	31.2	A
	CO-60	3.16E-02	6.91E-04	3.27E-02	1.04	45.7	A
	SR-85	5.03E-02	1.62E-03	5.27E-02	1.05	31.0	A
	Y-88	7.06E-02	1.84E-03	7.47E-02	1.06	38.4	A
	CD-109	6.60E-01	3.82E-02	7.12E-01	1.08	17.3	A
	SN-113	4.07E-02	1.59E-03	4.33E-02	1.06	25.6	A
	CS-137	1.98E-02	6.07E-04	2.01E-02	1.01	32.7	A
	CE-137	2.15E-02	3.71E-03	2.26E-02	1.05	5.8	A
	HG-203	4.66E-02	2.16E-03	4.70E-02	1.01	21.6	A
AIR FILTER DET.#1	CR-51	1.22E-02	1.04E-03	1.59E-02	1.30	11.8	A
	MN-54	1.96E-02	6.77E-04	1.95E-02	0.99	29.0	A
	CO-58	1.97E-03	1.56E-04	1.63E-03	0.83	12.6	A
	FE-59	4.91E-03	2.76E-04	< MDA		17.8	N
	CO-60	4.41E-02	9.72E-04	4.65E-02	1.05	45.3	A
	ZN-65	4.36E-02	1.27E-03	3.77E-02	0.86	34.3	A
	AM-241	4.72E-03	3.27E-04	5.15E-03	1.09	14.4	A

Table 1 (cont.)

SAMPLE	NUCLIDE	NRC VAL.[1]	NRC ERR.[1]	LIC.VAL.[1]	RATIO[2]	RES[3]	RESULT[4]
AIR FILTER DET.#3	CR-51	1.22E-02	1.04E-03	1.47E-02	1.21	11.8	A
	MN-54	1.96E-02	6.77E-04	1.88E-02	0.96	29.0	A
	CO-58	1.97E-03	1.56E-04	2.02E-03	1.02	12.6	A
	FE-59	4.91E-03	2.76E-04	3.97E-03	0.81	17.8	A
	CO-60	4.41E-02	9.72E-04	4.63E-02	1.05	45.3	A
	ZN-65	4.36E-02	1.27E-03	3.95E-02	0.91	34.3	A
	AM-241	4.72E-03	3.27E-04	4.88E-03	1.03	14.4	A
RCS DET.#1	NA-24	8.09E-05	2.73E-06	7.97E-05	0.99	29.6	A
	MN-56	7.11E-05	2.94E-06	6.31E-05	0.89	24.1	A
	CO-58	3.47E-06	3.45E-07	< MDA[5]		10.1	N
	CO-60	4.51E-06	2.29E-07	5.57E-06	1.23	19.7	A
	ZN-65	1.67E-04	5.07E-06	1.66E-04	1.00	32.9	A
	ZN-69M	7.39E-05	2.67E-06	7.51E-05	1.02	27.7	A
	AS-76	1.39E-04	4.10E-06	1.49E-04	1.07	34.0	A
	SR-91	1.22E-04	5.77E-06	1.18E-04	0.97	21.1	A
	SR-92	3.18E-04	8.58E-06	3.13E-04	0.98	37.1	A
	Y-92	2.26E-04	1.88E-05	2.56E-04	1.13	12.0	A
	MO-99	3.07E-05	3.21E-06	3.52E-05	1.15	9.6	A
	I-131	2.19E-06	3.40E-07	1.88E-06	0.86	6.4	A
	I-133	3.52E-05	1.07E-06	3.59E-05	1.02	32.9	A
	I-134	4.17E-04	1.43E-03	4.46E-04	1.07	0.3	A
	I-135	9.95E-05	3.48E-06	1.01E-04	1.01	28.6	A
	BA-139	4.05E-04	1.45E-04	4.35E-04	1.08	2.8	A
	BA-140	8.68E-06	1.47E-06	7.57E-06	0.87	5.9	A
RCS DET.#2	NA-24	8.09E-05	2.73E-06	8.43E-05	1.04	29.6	A
	MN-56	7.11E-05	2.94E-06	7.09E-05	1.00	24.1	A
	CO-58	3.47E-06	3.45E-07	< MDA		10.1	N
	CO-60	4.51E-06	2.29E-07	5.44E-06	1.21	19.7	A
	ZN-65	1.67E-04	5.07E-06	1.62E-04	0.97	32.9	A
	ZN-69M	7.39E-05	2.67E-06	7.89E-05	1.07	27.7	A
	AS-76	1.39E-04	4.10E-06	1.47E-04	1.06	34.0	A
	SR-91	1.22E-04	5.77E-06	1.18E-04	0.97	21.1	A
	SR-92	3.18E-04	8.58E-06	3.24E-04	1.02	37.1	A
	Y-92	2.26E-04	1.88E-05	2.15E-04	0.95	12.0	A
	MO-99	3.07E-05	3.21E-06	3.48E-05	1.13	9.6	A
	I-131	2.19E-06	3.40E-07	2.63E-06	1.20	6.4	A
	I-133	3.52E-05	1.07E-06	3.79E-05	1.08	32.9	A
	I-134	4.17E-04	1.43E-03	4.68E-04	1.12	0.3	A
	I-135	9.95E-05	3.48E-06	1.04E-04	1.05	28.6	A
	BA-139	4.05E-04	1.45E-04	4.55E-04	1.12	2.8	A
	BA-140	8.68E-06	1.47E-06	4.19E-06	0.48	5.9	*

Table 1 (cont.)

SAMPLE	NUCLIDE	NRC VAL.[1]	NRC ERR.[1]	LIC.VAL.[1]	RATIO[2]	RES[3]	RESULT[4]
CHAR DET.#2	I-131	2.30E-03	6.78E-05	2.23E-03	0.97	34.0	A
	I-133	5.20E-03	1.82E-04	5.50E-03	1.06	28.6	A
CHAR DET.#3	I-131	2.30E-03	6.78E-05	2.23E-03	0.97	34.0	A
	I-133	5.20E-03	1.82E-04	5.48E-03	1.05	28.6	A
RCS CRUD DET.#2	CR-51	1.18E-01	5.35E-03	1.22E-01	1.03	22.1	A
	MN-54	1.18E-01	5.35E-03	1.38E-01	1.16	22.1	A
	CO-58	1.91E-02	7.40E-04	2.00E-02	1.05	25.8	A
	FE-59	5.71E-02	1.59E-03	5.98E-02	1.05	35.9	A
	CO-60	1.14E-01	2.39E-03	1.17E-01	1.02	47.8	A
	CU-64	5.56E-01	1.21E-01	5.94E-01	1.07	4.6	A
	ZN-65	2.76E-01	7.70E-03	2.85E-01	1.03	35.8	A
	ZN-69M	5.45E-02	1.99E-03	5.89E-02	1.08	27.3	A
	AS-76	2.37E-02	1.96E-03	3.02E-02	1.27	12.1	A
	NB-95	2.02E-03	3.26E-04	2.38E-03	1.18	6.2	A
	ZR-95	3.26E-03	5.06E-04	2.87E-03	0.88	6.4	A
	ZR-97	1.08E-02	1.23E-03	1.09E-02	1.01	8.8	A
	SB-122	6.02E-03	6.50E-04	7.46E-03	1.24	9.3	A
	SB-124	5.22E-03	3.22E-04	5.17E-03	0.99	16.2	A
	W-187	1.49E-02	9.67E-03	1.65E-02	1.11	1.5	A
RCS CRUD DET.#3	CR-51	1.18E-01	5.35E-03	1.27E-01	1.08	22.1	A
	MN-54	1.18E-01	5.35E-03	1.45E-01	1.22	22.1	A
	CO-58	1.91E-02	7.40E-04	2.13E-02	1.12	25.8	A
	FE-59	5.71E-02	1.59E-03	6.08E-02	1.06	35.9	A
	CO-60	1.14E-01	2.39E-03	1.21E-01	1.05	47.8	A
	CU-64	5.56E-01	1.21E-01	5.46E-01	0.98	4.6	A
	ZN-65	2.76E-01	7.70E-03	2.96E-01	1.07	35.8	A
	ZN-69M	5.45E-02	1.99E-03	6.02E-02	1.11	27.3	A
	AS-76	2.37E-02	1.96E-03	3.12E-02	1.32	12.1	A
	NB-95	2.02E-03	3.26E-04	2.49E-03	1.23	6.2	A
	ZR-95	3.26E-03	5.06E-04	1.87E-03	0.57	6.4	A
	ZR-97	1.08E-02	1.23E-03	1.07E-02	0.99	8.8	A
	SB-122	6.02E-03	6.50E-04	8.40E-03	1.40	9.3	A
	SB-124	5.22E-03	3.22E-04	5.48E-03	1.05	16.2	A
	W-187	1.49E-02	9.67E-03	1.65E-02	1.11	1.5	A
	NP-239	2.60E-03	1.10E-03	2.35E-03	0.90	2.4	A
GAS DET.#1	KR-85M	2.81E-04	1.52E-05	3.02E-04	1.08	18.4	A
	KR-87	1.84E-03	1.00E-04	2.01E-03	1.09	18.3	A
	KR-88	1.16E-03	6.30E-05	1.20E-03	1.03	18.4	A
	XE-133	8.72E-05	2.27E-05	1.27E-04	1.45	3.8	A
	XE-135	1.86E-03	3.39E-04	1.82E-03	0.98	5.5	A
	XE-135M	1.04E-02	2.05E-02	1.07E-02	1.03	0.5	A
	XE-138	3.93E-02	1.58E-03	4.06E-02	1.03	25.0	A

Table 1 (cont.)

SAMPLE	NUCLIDE	NRC VAL.[1]	NRC ERR.[1]	LIC.VAL.[1]	RATIO[2]	RES[3]	RESULT[4]
GAS DET.#3	KR-85M	2.81E-04	1.52E-05	3.24E-04	1.16	18.4	A
	KR-87	1.84E-03	1.00E-04	1.92E-03	1.05	18.3	A
	KR-88	1.16E-03	6.30E-05	1.17E-03	1.01	18.4	A
	XE-133	8.72E-05	2.27E-05	1.45E-04	1.66	3.8	A
	XE-135	1.86E-03	3.39E-04	1.88E-03	1.01	5.5	A
	XE-135M	1.04E-02	2.05E-02	9.64E-03	0.93	0.5	A
	XE-138	3.93E-02	1.58E-03	4.05E-02	1.03	25.0	A

1. These are dimensionless quantities and are compared on a relative basis only.
2. Ratio = Licensee Value / NRC Value
3. Resolution = NRC Value / NRC Error (one standard deviation)
4. Result : The result of the comparison is based on the criteria in Attachment 1 and is expressed by the following:

   A = Agreement   * = Criteria Relaxed
   D = Disagreement   N = No Comparison

5. Measured value was less than the licensee's minimum detectable activity.

ATTACHMENT 1

CRITERIA FOR COMPARING ANALYTICAL MEASUREMENTS

This attachment provides criteria for comparing results of capability tests and verification measurements. The criteria are based on an empirical relationship which combines prior experience and the accuracy needs of this program.

In these criteria, the judgment limits are variable in relation to comparisons of the NRC's value to its associated one sigma uncertainty. As that ratio, referred to in this program as "Resolution", increases, the acceptability of a licensee's measurement should be more selective. Conversely, poorer agreement should be considered acceptable as the resolution decreases. The values in the ratio criteria may be rounded to fewer significant figures reported by the NRC Reference Laboratory, unless such rounding will result in a narrowed category of acceptance.

RESOLUTION	RATIO = LICENSEE VALUE/ NRC REFERENCE VALUE AGREEMENT
< 4	NO COMPARISON
4 - 7	0.5 - 2.0
8 - 15	0.6 - 1.66
16 - 50	0.75 - 1.33
51 - 200	0.80 - 1.25
> 200	0.85 - 1.18

Some discrepancies may result from the use of different equipment, techniques, and for some specific nuclides. These may be factored into the acceptance criteria and identified on the data sheet.