U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-237/91032(DRS); 50-249/91035(DRS)

Docket Nos. 50-237; 50-249 Licenses No. DPR-19; DPR-25

EA-91-164

Licensee: Commonwealth Edison Company
Opus West III
1400 Opus Place
Downers Grove, IL 60515

Facility Name: Dresden Nuclear Power Station, Units 2 and 3

Inspection At: Dresden Site, Morris, IL

Inspection Conducted: October 28 through November 12, 1991

Inspectors: V. P. Lougheed 11/26/91 Date

G. M. Nejfelt 11/26/91 Date

Approved By: M. P. Phillips, Chief 11/27/91 Date
Operational Programs Section

Inspection Summary

Inspection on October 28 through November 12, 1991
(Reports No. 50-237/91032(DRS); 50-249/91035(DRS))

Areas Inspected: This was a special announced safety inspection by regional based inspectors to review the events surrounding the failure of containment purge valve 3-1601-24. A general review of the local leak rate test program was also performed. Inspection module 61720 was used during this inspection.

Results: The inspection resulted in one apparent violation against Technical Specification 3.7.D which requires that containment isolation valves be operable during periods of power operation. During the last operating cycle, valve 3-1601-24 was inoperable because, when the valve operator was fully closed, the valve disk was partially open. This apparent violation is

described in section 3 of the report. The report also discusses an unresolved item relating to an apparent 50.59 violation where a required Technical Specification update had not been made (paragraph 4.A). A licensee strength was identified in the conduct and attitudes of the test personnel, which resulted in improvements in the overall test program.

REPORT DETAILS

1. Persons Contacted

Commonwealth Edison

L. Gerner, Technical Superintendent
J. Kotowski, Production Superintendent
J. Gates, Assistant Technical Staff Supervisor
K. Peterman, Regulatory Assurance Supervisor
M. Andjelic, Local Leak Rate Test Coordinator
D. Booth, Master Electrician
B. Colebank, Post-Maintenance Testing Coordinator
R. Geier, Master Maintenance Mechanic
J. Harrington, Nuclear Quality Programs Maintenance Group Leader
M. Horbaczewski, Inservice Testing Group Leader
M. Korchynsky, Unit 3 Operating Engineer
D. Legler, Site Engineer
D. Lowenstein, Regulatory Assurance Analyst
R. Stachniak, Performance Improvement Supervisor
D. VenPelt, Assistant Maintenance Supervisor
G. Whitman, Inservice Inspection Coordinator
K. Yates, Onsite Nuclear Safety Administrator

U.S. NRC

W. Rogers, Senior Resident Inspector
D. Liao, Reactor Engineer

All of the above were present at the exit held November 12, 1991, with the exception of Mr. Horbaczewski.

The inspectors also interviewed other licensee employees during the course of the inspection.

2. Licensee Action on Previous Inspection Findings

A. (Closed) Unresolved Item 237/90017-05 "Components not in LLRT [Local Leak Rate Test] Program": This item was addressed in Inspection Reports No. 50-237/90006(DRS); No. 50-249/90005(DRS) and a non-cited violation was issued. Based on the evaluation and resolution contained in that report, this item is considered closed.

B. (Closed) Unresolved Item 237/90027-09 "Adequacy of Post-Maintenance Testing in Regard to Failure of Torus Purge Valve During an ILRT [Integrated Leak Rate Test]": This unresolved item was resolved through issuance of a violation. See following item for closure.

C. <u>(Closed) Violation 237/91006-01 "Inadequate Post-Maintenance Testing Resulting in Lack of Containment Integrity for an Entire Cycle"</u>: This violation, along with a proposed civil penalty, was issued in April 1991. The licensee acknowledged the validity of the violation and paid the civil penalty. Their response to the violation was acceptable. Therefore, this item is considered closed.

D. <u>(Closed) Systematic Evaluation Program Topics VI-4; VI-6 "Installation of Proper Leak Rate Test Taps on the Reactor Building Closed Cooling Water System"</u>: The inspectors reviewed the methodology used by the licensee to perform local leak rate testing of both the supply and return on the reactor building closed cooling water system. The inspectors were satisfied that the licensee was testing these penetrations in accordance with the requirements of Appendix J. This item is considered closed.

E. <u>(Open) Systematic Evaluation Program Topic VI-4, "Leakage Conditions Under Which the Remote Manual Isolation Valves on LPCI [Low Pressure Core Injection] and Core Spray Systems Should be Isolated Are Incorporated Into the Emergency Procedures"</u>: At the time of the inspection, these conditions were not covered in any emergency procedures, to the best of the licensee's knowledge.

The licensee was unable to discern who had immediate responsibility for closure of this item. Therefore, this item will remain open.

3. <u>Review of Events Surrounding the Failure of Unit 3 Containment Purge Valve 3-1601-24</u>

A. <u>Sequence of Events</u>

On December 25, 1989, a local leak rate test was performed on penetration X-125, the Drywell/Torus Vent, with successful results (8.49 standard cubic feet per hour (scfh)). This penetration was comprised of inside containment valves 3-1601-23 and 3-1601-62, and outside valves 3-1601-24, 3-1601-60, 3-1601-61, and 3-1601-63.

On January 27, 1990, a work request was generated to repair the operator on valve 3-1601-24, an 18-inch butterfly valve, due to failure of Dresden Technical Surveillance (DTS) 1600-27 "Fail-Safe Air Test." The maintenance performed involved replacement of the valve operator piston rod. A post-maintenance local leak rate test was neither specified nor performed. Maintenance was completed on February 3, 1990.

The following day (February 4, 1990) an integrated leak rate test was performed. The integrated test was acceptable in the as-left condition, with a final leakage rate of 1.02 wt%/day as compared to the allowable of 1.2 wt%/day).

Unit 3 was returned to power on February 11, 1990. At this time known leakage from all local leakage sources was 485.43 scfh. Technical specification allowable leakage was 488.452 scfh.

Approximately 19 months later, on September 9, 1991, Unit 3 shut down for its next refueling outage. A local leak rate test was performed on penetration X-125 on September 16, 1991. The penetration could not be pressurized. On September 20, 1991 the licensee identified that the leakage was through outboard isolation valve 3-1601-24. Investigation, by the licensee, into the cause of the failure determined that, as part of the maintenance performed during the previous cycle, a new piston rod had been installed. This piston rod increased the valve actuator stroke by approximately one-eighth of an inch which resulted in the valve disk rotating past the fully closed position to where it was partially reopened. The valve position had been incorrect over the entire previous operating cycle. After making this determination, the licensee immediately notified the NRC under the requirements of 10 CFR 50.72(b)(2)(iii).

B. <u>Regulatory Requirements</u>

Drywell purge valve 3-1601-24, a containment isolation valve, is addressed in Technical Specification Table 3.7.1. Technical Specification Limiting Condition for Operation 3.7.D required that, during power operation, all isolation valves listed in Table 3.7.1 were to be operable. If a containment isolation valve was not operable, then the unit was either to be shutdown, or the redundant isolation valves in the line were to be isolated, with their positions recorded daily. Valve 3-1601-24 could not be considered operable during the previous reactor power operating cycle, because with the operator in the fully closed position, the valve disk was partially open. Because the licensee did not recognize that the valve was inoperable, the limiting conditions for operation, specified in Technical Specification 3.7.D, were not met. This is an apparent violation of Technical Specification 3.7.D (249/91035-01(DRS)).

C. Root Cause

The root cause was characterized by the licensee, in Licensee Event Report (LER) 50-249/91-009, Revision 0, as "... inadequate controls were provided with the work package...." The inspectors concluded that the use of informal communication methods, due to inadequate procedures, contributed to the inadequate post-maintenance testing.

A local leak rate test was not specified as a required post-maintenance test following the valve operator maintenance. The maintenance workers did not convey back to the work analyst that they had installed a new piston rod, nor did the work analyst inform the technical staff of the known scope of the repair. The technical staff considered a local leak rate test following maintenance unnecessary, if the work performed was on only the valve operator, and not on the valve seating surface. As this was their understanding of the maintenance being performed on valve 3-1601-24, they did not require a local leak rate test.

The licensee identified, in their licensee event report, two other Unit 2 valves which had not had local leak rate test performed following operator maintenance. These were the reactor head cooling check valve, 2-205-27, and a torus/drywell purge outboard isolation valve, 2-1601-63. The inspectors concluded that a common cause, contributing to the failure to perform post-maintenance local leak rate tests on these valves, was that these were unanticipated maintenance activities which came up during the last two weeks of the outage. The group responsible for performing local leak rate tests did not have the opportunity to determine the scope of the maintenance and to identify to the work analysts that local leak rate tests were required, due to the informal methods of communication being used.

During the past two and a half years, the Quality Assurance organization (QA), in their audits and surveillances, neither considered whether appropriate post-maintenance tests were specified nor if the work analyst had sufficient information to determine what the appropriate post-maintenance tests should have been. Rather, QA only verified that the testing that was specified was that which was performed. Consequently, QA did not provide a comprehensive oversight of the post-maintenance testing program.

D. Corrective Actions

As part of their response to the event, the licensee immediately initiated a comprehensive audit to ensure that all containment isolation valves on the operating Unit 2 had local leak rate tests administered following maintenance. Additionally, those corrective actions being implemented due to the failure of the inner flange on Unit 2 torus purge valve, such as the development and distribution of color coded drawings showing the containment boundaries and isolation valves, were reviewed and expanded. Finally the lead engineer for the local leak rate test program was designated to review, on a daily basis, all work requests being performed. However, the inspectors identified that all of these corrective actions were being done informally rather than through formal procedures. For example, the initiative to perform the daily reviews, discussed above, was specified by a maintenance memorandum.

In the long term, the licensee committed to implement further corrective actions, including: (1) performance of a comprehensive review on Unit 3 to ensure no local leak rate tests were omitted during the current outage prior to Unit 3 startup; (2) development of a matrix identifying all components having local leak rate test, inservice test, or other mandatory post-maintenance testing requirements; (3) training, both initial and requalification, of staff and operators concerning local leak rate testing requirements; (4) formalization of post-maintenance testing requirements into a procedure; and (5) evaluation of approaches used by other Commonwealth Edison nuclear stations to ensure that appropriate post-maintenance testing requirements are included in work packages.

The inspectors had no concerns with the proposed long-term corrective actions.

4. Review of Local Leak Rate Test Program

A. Procedural Review

The licensee controlled the local leak rate test program through Dresden Administrative Procedure (DAP) 14-05, "Leak Rate Testing Program", Revision 5. Local leak rate testing of individual Type B and C components was done in accordance with the following Dresden Technical Surveillances:

1) (DTS) 1600-1, Revision 14 (primary containment valves);

2) DTS 1600-2, Revision 6 (Bellows); DTS 1600-4, Revision 9 (Electrical Penetrations);

3) DTS 1600-14, Revision 9 (Personnel Access Lock);

4) DTS 1600-15, Revision 8 (Double Gasketed Seals); and

5) DTS 0250-01, Revision 9, and DTS 0250-03, Revision 3, (Main Steam Isolation Valves Dry and Wet Tests).

The inspectors reviewed these procedures against the requirements of Appendix J, and the licensee's Technical Specifications.

The inspectors found that the procedures met all Appendix J testing requirements, however, two concerns were identified. The first involved the licensee's use of long lengths of tubing to pressurize penetrations. This was done to reduce the accumulated dose to the staff performing the tests. The inspectors discussed with the licensee the potential for a pressure drop through the tubing, as well as the methodology used by other sites to prevent this problem. The licensee committed to resolving this concern. The second concern related to those tests performed against reactor water head pressure. The licensee committed to review the procedures for performing tests against a water head, to ensure that these tests would be performed against the correct test pressure.

B. <u>Technical Specification/Appendix J Disagreement</u>

The inspectors identified that the licensee had changed their airlock test methodology because of a 1982 NRR Safety Evaluation Report which found that the licensee's previous method of testing the airlocks did not meet Appendix J requirements. When the licensee revised the airlock test to be consistent with Appendix J requirements, they failed to amend their Technical Specifications. As of November 1991, the Technical Specifications still specified that the airlock be tested at 10 psig with an acceptance criteria of 0.0375 La. The licensee was performing airlock testing at Pa, 48 psig, with an acceptance criteria of 0.05La.

The licensee had prepared a Technical Specification update; however, it had never been submitted to the Commission. This concern is being tracked as an Unresolved Item (237/91032-03(DRS); 249/91035-03(DRS)), pending further review of the circumstances surrounding the licensee's failure to amend the technical specifications at the time of changing its testing requirements.

C. Review of Testing Results

The inspectors reviewed the results of the as-found local leak rate tests for the 1991 Unit 3 refueling. The inspectors noted that, besides the X-125 penetration discussed above, the licensee had a number of other penetrations which also could not be pressurized. One of these was the single valve pathway on the high pressure coolant injection (HPCI) turbine exhaust. As a result of this one valves' failure, the containment exceeded both the 0.6La maximum pathway leakage requirement and the 0.75 minimum pathway leakage requirement. The licensee considered the large number of failed valves to be unusual and not in keeping with normal site practices.

D. Bellows

Because of a problem identified concerning the adequacy of Type B testing of bellows seal penetrations at Quad Cities, the licensee recognized that the local leak rate test results on Dresden Unit 3 might be non-conservative. The licensee performed tests on all the Unit 3 bellows with both air and helium. Only one penetration showed signs of excessive leakage (main steam line A) and was replaced during the outage. The licensee planned to submit a temporary waiver of compliance on the bellows to NRR prior to unit startup, explaining their intended course of action. Additionally, the licensee committed to perform an integrated leak rate test every refueling outage, on both units, until either the bellows had been replaced with testable bellows, or such testing was no longer necessary.

E. Licensee Self-Initiated Study of Containment Isolation Valves

The licensee initiated a study of all containment isolation valves on Dresden in the 1989-1990 time period. The licensee has committed to submitting the results of this study as a revision to the Updated Safety Analysis Report (USAR) Table 5.2.2.5 (i.e., the table would be updated to show which valves and penetrations required Type B or C testing). Several penetrations were added to the local leak rate testing program as a result of the study. However, the study also identified some penetrations where all the containment boundaries were not being properly tested (e.g., the inboard flanges on the drywell and torus purge valves). At the time of the inspection, the licensee stated that the study was on hold, awaiting a meeting to be held between the licensee and NRR. The date for that meeting had not been set.

5. Unresolved Items

Unresolved items were matters about which more information is required in order to ascertain whether they are acceptable items, items of noncompliance, or deviations. An unresolved item disclosed during the inspection is discussed in paragraph 4.B.

6. Exit Interview

The inspectors met with licensee representatives (denoted in paragraph 1) throughout the inspection. An exit meeting was held prior to leaving the site on November 12, 1991. During the exit, the inspectors summarized the scope and apparent findings of the inspection. The inspectors also discussed the likely informational content of the inspection report with regards to documents or processes reviewed by the inspectors during the inspection. The licensee did not identify any such documents or processes as proprietary.

test was not performed following the maintenance. The failure to prescribe and perform a local leak rate test as part of the post-maintenance testing was partially due to the use of informal communication between the maintenance groups and the local leak rate test group as to when these tests were required.

Accordingly, no Notice of Violation is presently being issued for this inspection finding. Please be advised that the number and characterization of apparent violations described in the enclosed inspection report may change as a result of further NRC review.

You will be advised by separate correspondence of the results of our deliberations on this matter. No response regarding the apparent violation is required at this time.

We do request, however, that you respond within 60 days of the date of this letter to the unresolved item identified in the attached inspection report. Your response should include the date when you expect to submit the Technical Specification amendment.

In accordance with 10 CFR 2.790 of the Commission's regulations, a copy of this letter and the enclosures will be placed in the NRC Public Document Room.

The responses directed by this letter are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, PL 96-511.

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

Sincerely,

Hubert J. Miller, Director
Division of Reactor Safety

Enclosure: Inspection Reports
No. 50-237/91032(DRS);
No. 50-249/91035(DRS)

<u>See Attached Distribution</u>

RIII	RIII	RIII	RIII
--------SEE PREVIOUS PAGE--------			
Lougheed	Nejfelt	Phillips	Wright
11/ /91	11/ /91	11/ /91	11/ /91
RIII	RIII	RIII	RIII
Burgess	for Pederson	Martin	Miller
11/27/91	11/[illegible]/91	11/[illegible]/91	11/[illegible]/91

Yes

Yes