U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report Nos. 50-237/90-022(DRP); 50-249/90-022(DRP)

Docket Nos. 50-237; 50-249 License Nos. DPR-19; DPR-25

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, Illinois

Inspection Conducted: June 28 through September 20, 1990

Inspector: D. E. Hills

Approved By: J. M. Hinds, Chief
Reactor Projects Section 1B

10-4-90
Date

Inspection Summary

Inspection during the period of June 28 through September 20, 1990 (Report Nos. 50-237/90-022(DRP); 50-249/90-022(DRP))
Areas Inspected: Special, announced safety inspection of the licensee's previous practice of utilizing a temporary sample pump to obtain the daily drywell air sample. (Module 92701)
Results: The inspection resulted in the identification of one apparent 10 CFR 50.59 violation in that the licensee's practice effectively constituted a change in Technical Specifications and an unreviewed safety question existed in regard to the temporary sample pump. Prior NRC approval was not sought or obtained. This temporary alteration reduced the margin of safety as defined in the basis of Technical Specifications in regard to the maximum allowable primary containment accident leak rate. Primary containment was effectively degraded to unacceptable levels whenever the daily drywell air samples were being obtained with the temporary sample pump. (50-237/90-022-01(DRP); 249/90-022-01(DRP), paragraph 4).

The apparent violation reflects adversely on the safety assessment/quality verification and engineering/technical support functional areas. It represents a significant failure to meet the requirements of 10 CFR 50.59 requirements in that a required license amendment was not sought prior to implementing a facility change. This effectively circumvented the NRC's role in the regulatory process. Analyses can be performed which may show, using more realistic assumptions than the more conservative assumptions contained in the plant licensing basis, that offsite and control room dose projections are

within acceptable criteria. However, the determination of the safety significance of the change (whether the change is safe or unsafe) for an unreviewed safety question is clearly an NRC function and not within the authority of the licensee.

DETAILS

1. Persons Contacted

Commonwealth Edison Company

*E. Eenigenburg, Station Manager
*L. Gerner, Technical Superintendent
D. Van Pelt, Assistant Superintendent - Maintenance
J. Kotowski, Production Superintendent
J. Achterberg, Assistant Superintendent - Work Planning
G. Smith, Assistant Superintendent-Operations
*K. Peterman, Regulatory Assurance Supervisor
M. Korchynsky, Operating Engineer
B. Zank, Operating Engineer
J. Williams, Operating Engineer
*M. Strait, Technical Staff Supervisor
L. Johnson, Q.C. Supervisor
D. Morey, Chemistry Services Supervisor
D. Saccomando, Health Physics Services Supervisor

The inspector also talked with and interviewed several other licensee employees, including members of the technical and engineering staffs.

*Denotes those attending one or more exit interviews conducted informally at various times throughout the inspection period.

2. Licensee Actions on Previously Identified Items (92701)

(Closed) Unresolved Item (50-237/90017-04(DRP)). This item concerned the licensee's past practice of utilizing a temporary sample pump to obtain daily drywell air samples. This action created an unattended and unmonitored vent path from the drywell (primary containment) through the sample line to the reactor building (secondary containment). This item was open pending completion of a licensee 10 CFR 50.59 safety evaluation regarding this practice. The licensee's safety evaluation was completed and is discussed in paragraph 5 of this report. This item also concerned the adequacy of the drywell manifold sample system containment isolation provisions. Further review indicated that the containment isolation provisions for this system were approved by the NRC in a Safety Evaluation Report (SER) dated March 5, 1980, in regard to NUREG-0578 Category A Item 2.1.4 (NUREG-0737 Item II.E.4.2) "Containment Isolation." As such, the inspector has no further concerns regarding this portion of the item. The review of licensee actions in regard to the temporary sample pump usage and the resulting affect upon primary containment indicated an apparent violation of 10 CFR 50.59 as discussed in paragraph 4 of this report. Since the apparent violation involving the temporary sample pump will be tracked as a separate item and the NRC previously approved the containment isolation provisions for the drywell manifold sample system, this unresolved item is considered closed.

3. Background

a. Drywell Manifold Sample System Description

The purpose of the drywell manifold sample system is to provide air samples to identify the location of reactor coolant pressure boundary leaks inside of the drywell. The drywell manifold sample system (one for each unit) is designed to take a suction from 22 sample points in primary containment with each half inch sample line having its own two manual primary containment isolation valves (both located outside of primary containment) and a filter cartridge. Flow then passes through a common header from which the sample pump takes a suction. Return back to the primary containment is provided through a connection to the continuous oxygen monitoring system which discharges to the drywell through two automatic containment isolation valves which close on a Group II isolation signal. Thus, the drywell manifold sampling system has automatic isolation only on its discharge. The containment isolation provisions for this system were approved by the NRC in a SER dated March 5, 1980 in regard to NUREG-0578 Category A Item 2.1.4 (NUREG-0737 Item II.E.4.2), "Containment Isolation."

b. Daily Drywell Air Samples Utilizing Temporary Sample Pump

Since 1978 and possibly before, the licensee used a temporary sample pump as a backup method to obtain the Technical Specification required daily drywell air sample. Use of the temporary sample pump was frequent, especially in the last couple of years due to recurring problems with the permanent pumps. The licensee indicated that the permanent pumps were operable only a few weeks through the major portion of 1988 through 1990. The licensee also indicated that the reliability of these pumps was poor prior to 1988. Use of the temporary sample pump involved breaking the closed loop on the drywell manifold sample system below the sample filter on one of the sample lines, attaching a rubber hose with a quick disconnect fitting, connecting the hose to the temporary sample pump and discharging the pump exhaust to the reactor building. The system was left unattended while a sample was being taken although automatic isolation was not provided. Obtaining a representative sample required running the system in this configuration for at least 50 minutes. (A subsequent procedure specified a minimum of one hour.) This allowed an unattended and unmonitored path from the drywell (primary containment) through the sample line to the reactor building (secondary containment). A procedure was written on May 25, 1989 to cover this operation due to a nondocumented third party reviewer comment. This procedure contained a prerequisite to notify the control room prior to sampling and a precaution that the two valves upstream of each filter holder must be closed when drywell isolation is required. No analysis was done by the licensee to determine the effect on the offsite and control room doses in

consideration of manual reaction time and accessibility during design basis accidents. The licensee's technical staff system engineer identified the problem on June 28, 1990.

c. Ramifications of Temporary Sample Pump Usage

This use of the temporary sample pump in this configuration was contrary to Technical Specification 3.7.A.2 which required primary containment integrity when the reactor was critical or the reactor water temperature was above 212 degrees F. (The definition of primary containment integrity requires that all manual isolation valves on lines connecting to containment which are not required to be open during accident conditions are closed.) In this configuration, at a postulated design basis loss of coolant accident (LOCA) value of 48 psig, the licensee determined that a leakage of 4.73 percent per day would occur through this line. This exceeded the Technical Specification allowed primary containment leakage test value of 1.6 percent per day. (The Technical Specification limit would actually be exceeded by a greater amount when leakage from this line is added to other leakage sources.) The applicable Technical Specification action statement 3.0.A required hot shutdown within 12 hours and cold shutdown within the following 24 hours. Since the isolation valves were open for sampling for a sufficiently short duration, this action statement was not exceeded.

4. Comparison of Practice to Requirement

10 CFR 50.59 states that a holder of a license may (i) make changes in the facility as described in the safety analysis report (SAR), (ii) make changes in the procedures as described in the safety analysis report, and (iii) conduct tests or experiments not described in the safety analysis report, without Commission approval, unless the proposed change, test or experiment involves a change in the technical specifications incorporated in the license or an unreviewed safety question. The licensee on numerous occasions since at least 1978 and without NRC approval made changes in the facility as described in the safety analysis report, which involved a change in Technical Specifications and constituted an unreviewed safety question, by performing a temporary alteration utilizing a temporary sample pump to obtain the daily drywell air sample. This temporary alteration reduced the margin of safety as defined in the basis of Technical Specifications in regard to the maximum allowable primary containment accident leak rate. This is an apparent violation (50-237/90-022-01(DRP); 50-249/90022-01(DRP)).

The temporary alteration represented a change in the facility as described in various portions of the SAR in regards to the drywell manifold sample system design, the primary containment leak rate and its affect on the accident analysis.

Technical Specification 3.7.A.2.a.(3) prescribes a maximum allowable test leakage rate of 1.6 percent by weight of the containment air per 24 hours at 48 psig. Usage of the temporary sample pump with the single line represented an additional leakage of 4.73 percent per day beyond normal containment leakage. This number exceeds the technical specification allowable leakage and therefore effectively constitutes a change in the Technical Specifications incorporated in the license.

10 CFR 50.59 states that "a proposed change, test, or experiment shall be deemed to involve an unreviewed safety question (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced".

Technical Specification 4.7.A basis indicates that the design basis loss of coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 2.0 percent per day at 48 psig. This was the basis for determining a maximum allowable test leak rate of 1.6 percent per day at a pressure of 48 psig. (The difference was to account for the effects of containment environment under accident and test conditions by applying a 0.8 correction factor.) Technical specification 4.7.A basis states that "the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected offsite doses and 10 CFR 100 guidelines." It further states that "although the dose calculations suggest that the accident leak rate could be allowed to increase to about 3.2 percent per day before the guideline thyroid doses given in 10 CFR 100 would be exceeded, establishing the test limit of 1.6 percent per day provides adequate margin of safety to assure the health and safety of the general public." Usage of the temporary sample pump represented an additional leakage of 4.73 percent per day during accident conditions. This additional leakage rate nullified the margin of safety as defined in the Technical Specification basis and also exceeded the value the Technical Specification basis indicates as an acceptable consequence to public health and safety under 10 CFR 100 if a design basis LOCA occurred during sampling.

Although there have been subsequent licensing actions regarding the 10 CFR 100 analysis for the design basis loss of coolant accident, the licensee was not able to provide any documentation indicating that the margin of safety defined in the Technical Specification basis was specifically changed.

5. Comparison of Licensee Safety Evaluation and Supporting Analysis With Licensing Bases and Technical Specifications

Following identification of this issue, the licensee performed a 10 CFR 50.59 safety evaluation for unattended usage of the temporary sample pump to determine whether an unreviewed safety question existed. The licensee's analysis indicated that requirements were met and the safety evaluation indicated that an unreviewed safety question regarding past usage of the temporary sample pump did not exist.

NRC review of the licensee's analysis regarding this issue in comparison to licensing basis assumptions indicates questionable rationale with the licensee's conclusions. (The NRC review considered only assumptions used in the analysis and not the methodology and computer codes for the calculations themselves). In particular, in order to show acceptable values, this analysis used assumptions that were contrary to more conservative assumptions specifically stated in the licensing basis, and, in some cases, reflected in Technical Specifications. Even with the open line (an additional 4.73 percent per day leakage), the licensee's analysis showed significantly smaller offsite doses than the December 7, 1981 SER for Systematic Evaluation Program (SEP) topic XV-19 "Loss of Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks With in the Reactor Coolant Pressure Boundary." These reductions may have been in part accomplished by applying specific assumptions used in the control room dose analysis (and subsequently approved in an SER for that analysis) to the offsite dose analysis. Although the permissibility of this is unclear, there may be some merit to this approach, as long as the applicability is essentially the same and there are not technical reasons to prevent on a specific case (such as inconsistencies with Technical Specifications), the usage of less conservative assumptions approved for another type of analysis.

However, some of the licensee's assumptions were less conservative than the assumptions used in the licensing basis for the control room dose analysis (SER on the control room habitability study) or Technical Specification values. The control room habitability SER was issued on May 11, 1983 for NUREG-0737, Item III.D.3.4, Control Room Habitability, which accepted the licensee's control room habitability study as indicating an acceptable control room ventilation design. In order to achieve acceptable results for the temporary sample pump analysis, the licensee had to assume at least a Standby Gas Treatment System (SGTS) organic efficiency of 96.3 percent. However, the Technical Specification acceptance criteria is 90 percent. Although the higher efficiency could probably be justified based upon historical testing results, this is still less conservative in regard to Technical Specifications and thus is not justifiable to use in a 10 CFR 50.59 safety evaluation. The licensee implicitly recognized this inconsistency when it re-performed the control room habitability study to incorporate the 90 percent value on April 19, 1988. The licensee's safety evaluation for the temporary sample pump issue assumed normal control room ventilation operation for only 40 minutes as opposed to the eight hours specified in the licensing basis. The licensee's reanalysis for the control room habitability study made the same 40 minute assumption. The licensee, contrary to the Technical Specification basis, assumed a normal containment leakage rate of 1.6 percent per day instead of the apparently required 2.0 percent per day rate. Although Technical Specifications prescribed a 1.6 percent per day maximum allowable test leak rate, the Technical Specification basis indicated the actual maximum allowable accident leak rate was 2.0 percent per day. (The testing acceptance criteria were derived from this value by applying a correction factor to account for uncertainties from the effects of the testing environment compared to the accident environment.) Therefore, usage of the smaller value in offsite and control room dose calculations was non-conservative. Previous NRC

10 CFR 100 calculations for offsite doses reflected in the original licensing of the plant and the SER for SEP topic XV-19 used 2.0 percent per day leakage. Both the control room habitability analysis and reanalysis used 1.6 percent per day leakage. The acceptability of these last two assumptions, specifically for control room habitability, are an unresolved item (237/90-022-02 (DRP); 249/90-022-02 (DRP). Finally, the analysis assumed a constant maximum design accident pressure of 48 psig over the entire course of the accident in accordance with the licensing basis but assumed a decreasing accident pressure for the extra 4.73 percent per day leakage portion. This last assumption was not only contrary to the licensing basis but also contrary to the current Standard Review Plan (SRP) provisions which prescribe a constant maximum pressure for built-in conservatism to the calculations.

Following NRC and licensee discussions of these issues with respect to the safety evaluation, the licensee provided additional clarification of the intent of the safety evaluation. This clarification acknowledged that assumptions used in analyses supporting forward looking 10 CFR 50.59 safety evaluations should be consistent with those previously approved by the NRC. However, since this analysis was performed for a past practice that was discontinued following discovery, the licensee believed these assumptions were appropriate to indicate whether an unreviewed safety question existed.

In summary, the licensee's analysis concerning this issue resorted to non-conservative assumptions with respect to the plant licensing basis or Technical Specifications in order to achieve acceptable results. Review of the licensee's analysis indicates that the calculated consequences of an accident may have been increased if the initial assumptions were in accordance with Technical Specifications and the licensing basis. All acceptance criteria exceeded cannot be explicitly identified without a reanalysis using the more conservative assumptions.

The licensee's safety evaluation for this issue indicated that the margin of safety as defined in Technical Specifications was not reduced based upon the definition of margin of safety as defined in Nuclear Safety Analysis Center (NSAC)-125 "Guidelines for 10 CFR 50.59 Safety Evaluation." This was due to the resulting doses being less than the acceptance limit (10 CFR 100 and General Design Criterion 19) in the licensing basis. This evaluation indicated a lack of understanding of NSAC-125 provisions. NSAC-125, Section 3.6 states that "changes in barrier performance that do not result in increased radiological dose to the public are addressed under margin of safety." NSAC-125 indicates an increase in consequences of accidents must involve an increase in doses above the licensing limit; however, the margin of safety as defined in the basis of any Technical Specification does not rely on this provision. Therefore, the dose to the public is not the determining factor in the margin of safety portion of the definition of an unreviewed safety question. In actuality, the margin of safety was reduced whether a comparison against 10 CFR 50.59 wording is used or NSAC-125 provisions are relied upon.

In addition, the licensee's safety evaluation also stated that the dose analysis methodology was based upon the original FSAR and SER dated August 31, 1966. However, the SERs which covered the final licensing of the plant dated October 17, 1969 for Unit 2 and November 18, 1970 for Unit 3 granted approval based on different calculations than the previous SER. The licensee could not locate any copies of these other SERs indicating a failure to maintain knowledge of the licensing basis. The inspector subsequently obtained copies of these SERs and provided them to the licensee. Licensee deficiencies in incorporating SERs in the Updated Final Safety Analysis Report is an unresolved item (237/90-022-03; 249/90-022-03 (DRP)).

6. Corrective Actions

As a result of this problem, the licensee completed or is planning the following actions:

a. A preliminary analysis was performed to quantify the amount of leakage through a one half inch primary containment penetration at design accident pressure. After finding that this greatly exceeded allowable limits the licensee reported the problem, in accordance with 10 CFR 50.72 and 50.73.

b. A temporary change to the procedure regarding usage of the temporary sample pumps was issued to require an individual in continual attendance and in contact with the control room by radio while the manual isolation valves are open. A temporary alteration was subsequently performed that moved the sample point for the Technical Specification required daily sample to a line that had automatic isolation. The temporary procedure change was discontinued following the temporary alteration.

c. All Radiation Protection shift personnel were briefed as to the problem to preclude improper usage of the system.

d. A deviation report was initiated to track the licensee's investigation of the problem. A potentially significant event report was also initiated for corporate management.

e. The licensee performed a safety evaluation on the unattended usage of the temporary sample pump which indicated that an unreviewed safety question did not exist.

f. The licensee is currently reviewing the problems with the permanent sample pumps and assessing what is needed to complete repairs.

g. A review of the design basis and the need for any system design improvements is being conducted. Since the design basis could not be identified, the licensee decided to reconstitute the design basis. The licensee is reviewing whether the system will be repaired and used or whether it is to be abandoned, dismantled and the lines capped.

h. A review is being conducted to determine possible methods whereby a temporary return line to the drywell could be established for use with the temporary sample pump. (Although automatic isolation is now provided, the temporary sample pump still exhausts to the reactor building which presents ALARA considerations.)

i. The licensee is reviewing other practices, procedures and surveillances for any other items that could violate containment integrity or system operability.

7. Root Cause

The root cause is attributed to a management deficiency in that safety evaluation administrative requirements were inadequate or were not adequately applied. As administrative requirements were upgraded over the years, no actions were taken to ensure past standing practices were reviewed.

A 10 CFR 50.59 safety evaluation was never done on this alteration (use of the temporary sample pump) since the original administrative requirements for temporary alterations only applied to lifted leads and jumpers. When the administrative requirements expanded to mechanical equipment, previously existing alterations were not evaluated for applicability to the administrative requirements. As such, in recent years each time this alteration was performed it was done contrary to the licensee's administrative procedures. (Dresden Administrative Procedure (DAP) 7-4 "Control of Jumpers or Lifted Leads," Revision 8, was issued on December 24, 1985, which added the requirement for 10 CFR 50.59 safety evaluations to be performed on jumpers and lifted leads. Revision 11 of this procedure was issued on August 15, 1988 to expand the definition of temporary alteration to additional items such as mechanical equipment.)

A procedure covering the use of the temporary sample pump did not exist until 1989 and thus the problem was not previously discovered through a procedure safety evaluation. Due to a non-documented third party reviewer's comment concerning use of the temporary sample pump without a procedure, Dresden Radiation Protection (DRP) Procedure 1350-3, "Sampling the Drywell Manifold System Using the Radeco Air Sampler" was first issued in May 1989. This was a missed chance to detect the problem since a 10 CFR 50.59 safety evaluation should have been performed. The screening criteria in effect at the time allowed entire categories of procedures (such as DRPs not related to effluent monitoring) to be automatically ruled out for a safety evaluation as long as they were not new or changed procedures or administrative controls described in the Final Safety Analysis Report (FSAR) or Technical Specifications. In this particular case, since it was a new procedure, the criteria required a safety evaluation to be performed. However, the reviewers mistakenly used the wrong administrative path as if it were a revision to this type of procedure instead of a new procedure. Therefore, a safety evaluation was not performed due to a failure to follow the administrative requirements. Additionally, the criteria themselves were still inappropriate since the licensee could have instead just made a revision to DRP 1350-7, "Operation of the Unit 2(3) Drywell Air Sampling Manifold

System" to allow usage of the temporary sample pump. In that case, the licensee's administrative requirements would not have required a safety evaluation to be performed with the same result (usage of the temporary sample pump without a safety evaluation). The screening criteria were revised on January 25, 1990, such that this is no longer a concern for recently issued procedures and revisions.

8. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, violations, or deviations. The unresolved items disclosed during the inspection are discussed in paragraph 5.

9. Exit Interview (30703)

The inspectors summarized the scope and findings of the inspection by telephone with the licensee's representatives (denoted in paragraph 1) on September 27, 1990. The licensee acknowledged this information. The inspector also discussed the likely informational content of the inspection report with regard to documents or processes reviewed by the inspector during the inspection. The licensee did not identify any such documents/processes as proprietary.