U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-341/87048(DRP)

Docket No. 50-341 Operating License No. NPF-43

Licensee: Detroit Edison Company
2000 Second Avenue
Detroit, MI 48226

Facility Name: Fermi 2

Inspection At: Fermi Site, Newport, Michigan

Inspection Conducted: October 18, 1987 through March 31, 1988

Inspectors: W. G. Rogers, Senior Resident Inspector
M. E. Parker, Resident Inspector

Approved By: R. Cooper, Chief
Reactor Projects Section 3B

5/5/88
Date

Inspection Summary

Inspection on October 18, 1987 through March 31, 1988 (Report No. 50-341/87048(DRP))

Areas Inspected: Special unannounced inspection by a resident inspector of the events surrounding the failure of licensed operators to comply with the Technical Specification action requirements associated with a reactor protection system instrument channel and of the isolation design configuration of the primary containment radiation monitor.
Results: Three violations were identified (failure to comply with a Technical Specification action statement, failure to provide adequate procedure content and inadequate primary containment isolation capability).

DETAILS

1. Persons Contacted

   a. Detroit Edison Company

      F. Abramson, Operations Engineer
      *D. Gipson, Plant Manager
      *P. Anthony, Compliance
      #*S. Catola, Vice President, Nuclear Engineering
      #*L. Goodman, Licensing Supervisor
      J. Green, Systems Engineering
      R. Laubenstein, Nuclear Assistant Shift Supervisor
      J. Leman, Director, Plant Safety, Nuclear Production
      *R. Lenart, General Director, Nuclear Engineering
      L. Lessor, Advisor to Plant Manager
      R. Lightfoot, Nuclear Shift Supervisor
      #*W. Orser, Vice President, Nuclear Operations/Plant Manager
      J. Plona, Operations Support Engineer
      E. Preston, Assistant Director, Plant Safety
      B. Sheffel, Nuclear Production, Technical Engineering ISI
      F. Svetkovich, Technical Engineer, Nuclear Production
      #*B. R. Sylvia, Group Vice President, Nuclear Operations
      # L. Wooden, Supervisor, I&C
      # L. Fron, Supervisor, Mechanical and Fluid Systems
      # P. Marquart, General Attorney
      *W. Tucker, Superintendent, Operations

   b. U.S. Nuclear Regulatory Commission

      *M. Parker, Resident Inspector
      # P. Pelke, Project Inspector
      #*W. Rogers, Senior Resident Inspector
      # C. Anderson, Enforcement Specialist
      # R. Cooper, Chief, Projects Section 3B
      # H. Wong, Sr. Enforcement Specialist, OE
      # Dr. C. J. Paperiello, Deputy Regional Administrator
      # R. Knop, Chief, Projects Branch 3
      # M. Virgilio, Deputy Director, DRP
      # T. Quay, Licensing Project Manager, NRR

   *Denotes those attending March 30, 1988 exit meeting.
   #Denotes those attending the April 28, 1988 enforcement conference.

2. Review of Drywell Pressure Surveillance Testing

   a. Background

      In Inspection Report 50-341/87002(DRP), a violation of a Limiting Condition for Operation (LCO) action statement associated with the high pressure coolant injection/reactor core isolation cooling (HPCI/RCIC) systems was identified. That violation was very similar to the LCO violation which occurred during this inspection period. The previous violation also involved performance of an I&C surveillance in which licensed operators failed to recognize the Technical Specification implications.

In response to the HPCI/RCIC violation, the licensee stated that the violation was attributable to a lack of understanding of plant conditions and a lack of an impact statement in the procedure. The licensee further stated that an I&C surveillance procedures improvement program was in place to upgrade the I&C surveillance procedures as part of the corrective steps that would be taken to avoid further violations. This effort included the addition of impact statements documenting the ramifications, legal and physical, of performing a particular surveillance test and to specify the plant operational conditions under which the test may be performed. The I&C surveillance improvement program was to be completed by January 31, 1988.

To prevent another violation before the completion of the I&C surveillance improvement program, the licensee began generating interim impact statements. The interim impact statements were being generated as a surveillance became due but were not formally incorporated into the procedure or approved by the Onsite Review Committee (OSRO). The interim impact statement would then be formalized into that particular surveillance procedure when the procedure was revised under the total I&C surveillance procedure improvement effort.

b. Limiting Condition for Operation

On October 24, 1987, the Nuclear Assistant Shift Supervisor (NASS) signed on Plant Operations Manual (POM) 44.020.015, Revision 3, "NSSS - Drywell Pressure, Division I, Channel A Response Time Test; C71-N650A and C71-N050A," to allow the instrument and control (I&C) repairman to perform a response time test of transmitter C71-N050A and master analog trip unit C71-N650A for the nuclear steam supply shutoff system (NSSSS) drywell pressure input. Transmitter C71-N050A is one of two Division I NSSSS high drywell pressure channels for the reactor protection system instrumentation and the isolation actuation instrumentation required by Technical Specification (T.S.) 3.3.1 and 3.3.2, respectively.

At 12:25 p.m. EDT, on October 24, 1987, the I&C repairman defeated the high drywell pressure instrumentation for channel A, as part of the response time test. This action rendered the channel inoperable and the licensee entered into the Technical Specification two-hour action statement. T.S. Table 3.3.1-1 and Table 3.3.2-1, Table Notation (a) allows a channel to be placed in an inoperable status for up to two hours for required surveillance testing without placing the trip system in the tripped condition provided at least one operable channel in the same trip system is monitoring that parameter.

At 2:25 p.m. EDT, the two hour Technical Specification allowance for having the drywell pressure channel inoperable was exceeded and went unnoticed by all personnel involved with the surveillance. It was not until 3:15 p.m. EDT, when the I&C repairman reported to the operating shift that they were having trouble with the surveillance, that personnel recognized a possible time constraint problem with the Technical Specifications. The drywell pressure transmitter was subsequently returned to operable status at 3:37 p.m. EDT. This resulted in the high drywell pressure channel being inoperable from 12:25 p.m. to 3:37 p.m., a total of three hours and twelve minutes.

This is considered an apparent violation of Technical Specification 3.3.1 and 3.3.2 for failing to place the drywell pressure Channel A in the tripped condition or to return the inoperable channel to operable status within the two hour period (50-341/87048-01(DRP)).

c. Inspector Followup

The inspector reviewed the procedures and discussed the event with the licensee personnel involved. From these reviews the inspector ascertained that:

(1) An interim impact statement had been generated for this surveillance procedure. However, in preparing the interim impact statement for incorporation into the surveillance test package, the Shift Operations Advisor (SOA) made an error in that he incorrectly assumed that Technical Specifications allowed the trip channel to be out of service for surveillance testing for three hours when Technical Specifications only allows the channel to be out of service for testing for two hours. This error was further compounded when the operations engineer concurred with this impact statement.

(2) The incorrect information in the non-OSRO approved interim impact statement was in contradiction to OSRO approved Procedure POM 44.020.015, Step 4.13.

(3) Even the incorrect time restraints in the interim impact statement were not adhered to since the channel was not placed in the tripped condition after the three hour time limit had expired.

(4) No formal means existed for tracking short-term LCOs. Procedure POM 21.000.18, "Out-of-Specification Log," does not require short-term LCOs to be placed in the out-of-specification log.

(5) I&C personnel were aware that response time testing has typically taken one shift to complete and they were aware that this particular response time test would take longer than the two-hour LCO limit, but this knowledge was not transmitted to the operating authority.

(6) During the surveillance, the licensee only performed Sections 6.1, 6.2 and 6.3 of POM 44.020.015. This consisted of performing the response time test for transmitter C71-N050A and master Analog Trip Unit C71-N650A. This would not have resulted in any isolations or actuations. Had this been fully understood prior to performing the surveillance, drywell pressure Channel A could have been placed in the tripped condition with no adverse consequences and full compliance with Technical Specifications would have been achieved.

d. Impact Statement Review

On February 17, 1988, the inspector reviewed POM 44.020.015 to ensure that the interim impact statement discussed previously had been corrected to adequately reflect Technical Specification requirements and had been formally incorporated into the procedure thereby receiving OSRO approval. During this review it was observed that the impact statement had been incorporated into the procedure and approved by OSRO. However, during incorporation of the impact statement into the procedure the licensee failed to incorporate the correct impact statement. Specifically, the impact statement still allowed the channel to be inoperable for up to three hours without placing the channel in the tripped condition instead of the two hours required by Technical Specifications 3.3.1 and 3.3.2.

Further review identified that the licensee had taken the same action with the associated drywell pressure response time procedures:

POM 44.020.016, Revision 20,	Drywell Pressure, Division II, Channel B, Response Time Test
POM 44.020.017, Revision 20,	Drywell Pressure, Division I, Channel C, Response Time Test
POM 44.020.018, Revision 20,	Drywell Pressure, Division II, Channel D, Response Time Test

This is considered an apparent violation (50-341/87048-02(DRP)) of Technical Specification 6.8.1.d for failing to properly implement the requirements of T.S. 3.3.1 and 3.3.2 into surveillance procedures.

e. Summary

Although this LCO violation was caused by inadequate communications between I&C and Operations, this appears to be indicative of a breakdown in the overall understanding and appreciation of Technical Specifications. These same elements were apparent in the 50-341/87002 violation and as such the corrective actions taken in response to that violation were inadequate to preclude the current violation. Also, this event pointed out the weaknesses in the licensee's system for not tracking short-term Limiting Conditions for Operation (LCOs).

3. Primary Containment Radiation Monitoring System (PCRM) Design Configuration

a. Background

The primary containment radiation monitoring (PCRM) system is configured in a parallel arrangement with the drywell hydrogen/oxygen sample panel. Both systems normally operate continuously during reactor operation and sample the drywell atmosphere from five zones through primary containment penetrations X-48a through X-48e. Each of these five penetrations has an air-operated remote manual isolation valve (T50-F401A, F402A, F403A, F404A, and F405A) and an associated local manual valve (T50-F033A, F034A, F035A, F036A, and F037A).

The original isolation design for the PCRM system and the drywell hydrogen/oxygen sampling system was an acceptable alternative to GDC 56 described in the FSAR Section 6.2.4. Containment isolation requirements were achieved using a single isolation valve (T50-F401A through F405A) and this was based on a closed system outside the containment. The basis of a single remote manual isolation valve is described in the UFSAR Table 6.2.2 and Section 6.2.4.2.2.3.2. This design assumed that the PCRM system would operate following a loss-of-coolant accident (LOCA) and the PCRM system would be in compliance with the closed system requirements.

In January 1984, the licensee determined that the PCRM system did not comply with the specific closed system requirements. Specifically, the system was not qualified for containment design pressure and problems were noted with the seismic and material certifications provided by the vendor. The licensee subsequently determined that the PCRM was a non-essential system following a LOCA and should be automatically isolated upon receipt of a LOCA signal. As such, in early 1984, the two automatic isolation valves (T50-F450 and F451) and the two manual isolation valves (T50-F063 and F064) were added to isolate the PCRM on a LOCA signal (high drywell pressure).

The automatic isolation valves were added to provide the isolation of the now non-essential PCRM system and was intended to return the system to that of a closed system configuration. The licensee believed this configuration was an acceptable alternative to GDC 56. This configuration resulted in providing two barriers in the event of a LOCA and failure of the PCRM boundary, the first barrier being the newly installed automatic isolation valves (T50-F450 and F451) and the second barrier being the remote manual isolation valves (T50-F401A through F405A).

b. Licensee Event Report (LER)

On October 17, 1987, during implementation of Engineering Design Package (EDP) 1786 on the primary containment radiation monitoring (PCRM) system, the primary containment isolation was questioned as to the adequacy of containment integrity. The isolation boundary utilized was two solenoid-operated valves (T50-F450 and F451) which are not primary containment isolation valves. The containment isolation valves for this penetration (X48a through X48e) are remote manual isolation valves (T50-F401A through F405A) and receive no automatic isolation signals. The primary containment isolation valves were open dur'ng implementation of this EDP. It was further determined that valves T50-F450 and F451 were not properly qualified to satisfy containment integrity requirements.

On October 17, 1987, at 8:50 p.m. EDT, the licensee determined this was a potential loss of the primary containment integrity. The licensee took immediate action to isolate the primary containment boundary. The Division I PCMS was subsequently shut down and isolated. The isolation consisted of closing primary containment isolation valves T50-F401A through F405A for the Division I PCMS. This resulted in the plant being in a seven days and 30 day Limiting Condition for Operation as a result of having the primary containment H2/02 monitoring system and radiation monitoring system, respectively, out of service. After investigation into the isolation, the licensee determined the inadequate isolation was a reportable event and on October 17, 1987, at 9:30 p.m. EDT the licensee made the applicable notifications per 10 CFR 50.72 for a primary containment integrity violation.

Local leak rate testing (LLRT) on T50-F450 and T50-F451 was satisfactorily completed on October 18, 1987, and the applicable out of service log was subsequently cleared for these valves. This allowed the licensee to isolate the PCMS radiation monitor utilizing T50-F450/F451 and open the designated containment isolation valves T50-F401A through F405A. This action allowed the H2/02 monitor to be placed in service and took the plant out of the seven day LCO. However, a 30 day LCO was still in place for not having the PCMS radiation monitor in service.

Discussions with the resident inspector concerning the configuration resulted in the inspector questioning the current design configuration of the PCRM system to meet 10 CFR 50, Appendix A, General Design Criterion (GDC) 56 requirements.

c. Exemption Request

On October 27, 1987, in DECO Letter No. NRC-87-0211, the licensee requested a temporary exemption from the requirements of 10 CFR 50, Appendix A, Criterion 56 (GDC 56), Primary Containment Isolation. This request was a result of review by the NRC in determining that the current design configuration, for the Division I primary containment monitoring system, did not meet the requirements of GDC 56. This exemption request along with other correspondence identified the licensee's proposed course of action to return the primary containment radiation monitoring system to service utilizing the current isolation design.

On November 13, 1987, the NRC granted to the licensee an exemption to General Design Criterion 56 of Appendix A to 10 CFR Part 50.

This exemption permitted postponement of full compliance with GDC 56 for the primary containment radiation monitoring isolation until startup following the planned local leak rate testing in March 1988. To support operation pending incorporation of modifications of the PCRM isolation, while the exemption is in effect, the licensee committed to upgrade the effectiveness of the isolation scheme as described in NRC letter, dated November 13, 1987. This action included treating valves T50-F450, F451, F040, F046, F063 and F064 as primary containment isolation valves, in a manner consistent with any other valve listed in Technical Specifications. The licensee also committed to revise the Emergency Operating Procedures and enhance operator training as an interim compensatory measure.

On January 29, 1988, the licensee submitted a proposed Technical Specification (License Amendment) change which results from modifications to bring the PCRM isolation design up to the standards set forth in GDC 56. On March 29, 1988, NRR issued Amendment 17 to the operating license in response to the January 29, 1988 letter.

d. Inspector Followup

In reviewing the PCRM design, the licensee was unable to find any correspondence accepting this configuration as an acceptable alternative to GDC 56. Review of the UFSAR identified that it had not been updated to reflect the current design configuration and categorized penetrations X48a through X48e as Engineered Safety Feature (ESF) or ESF-related system penetrations and that these penetrations are attached to a closed system. Technical Specifications (TS) had also not reflected this modification to include the additional automatic isolation valves (T50-F450 and F451). In addition, the licensee's program in 1984 to install these valves failed to properly maintain these valves in accordance with the applicable requirements of GDC 56, 10 CFR 50 Appendix J and other testing requirements (functional testing, logic testing, positive indicator checks, LLRT testing, etc.).

The PCRM containment isolation system design as described in UFSAR Section 6.2.4 does not reflect the as-built system as required by 10 CFR 50.34(b). The deviation between the UFSAR and the as-built system was not evaluated in accordance with 10 CFR 50.59. This is an apparent violation (50-341/87048-03(DRP)).

e. Enforcement Conference

An Enforcement Conference was held in the Region III Office on April 28, 1988 to discuss the PCRM containment isolation system design. No new information was provided.

4. Exit Interview (30703)

The inspectors met with licensee representatives (denoted in Paragraph 1) on March 30, 1988, and informally throughout the inspection period and summarized the scope and findings of the inspection activities. 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/processes as proprietary. The licensee acknowledged the findings of the inspection.