NOTICE OF VIOLATION
AND
PROPOSED IMPOSITION OF CIVIL PENALTIES

The Detroit Edison Company
Enrico Fermi Atomic Power Plant
Unit 2

Docket No. 50-341
License No. NPF-43
EA 88-104

During NRC inspections conducted on October 18, 1987 to March 31, 1988, and January 17 to April 28, 1988, 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 (1988), the Nuclear Regulatory Commission proposes to impose civil penalties pursuant to Section 234 of the Atomic Energy Act of 1954, as amended (Act), 42 U.S.C. 2282, and 10 CFR 2.205. The particular violations and associated civil penalties are set forth below:

A. 10 CFR Part 50, Appendix A, General Design Criterion 56 requires, in part, that each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves both inside and outside primary containment unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis.

Contrary to the above, as of October 17, 1987, the containment isolation configuration for the primary containment radiation monitoring (PCRM) system violated the requirements of General Design Criteria 56 in that containment isolation valves were not provided on the system lines both inside and outside primary containment and this configuration was not accepted on some other defined basis.

This is a Severity Level III violation (Supplement I).
Civil Penalty - $100,000.

B.1. With the unit in Modes 1, 2, or 3, Technical Specification Limiting Condition for Operation Action Statement 3.7.2.b.2 requires that if a Control Room Emergency Filtration System flowpath damper is inoperable for seven days, the unit be placed in HOT SHUTDOWN within 12 hours and COLD SHUTDOWN in the following 24 hours.

Technical Specification 1.25 defines a system, subsystem, train, component, or device to be OPERABLE or having OPERABILITY when it is capable of performing its specified functions and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication, or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support functions.

Contrary to the above, at 10:15 p.m. on January 21, 1988, with the unit in Mode 1, a Control Room Emergency Filtration System flowpath damper, which had been inoperable for seven days because the necessary attendant noninterruptible air compressor was out-of-service, was not returned to service nor was the unit placed in HOT SHUTDOWN within 12 hours and COLD SHUTDOWN in the following 24 hours.

B.2. With the unit in Modes 1, 2, and 3, Technical Specification Limiting Condition Action Statement 3.6.5.3.a.1 requires that if one Standby Gas Treatment subsystem is inoperable for 7 days the unit be placed in HOT SHUTDOWN within 12 hours and COLD SHUTDOWN in the following 24 hours.

Technical Specification 1.25 defines a system, subsystem, train, component, or device to be OPERABLE or having OPERABILITY when it is capable of performing its specified functions and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication, or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support functions.

Contrary to the above, at 10:15 p.m. on January 21, 1988, with the unit in Mode 1, the Division II subsystem of Standby Gas Treatment, which had been inoperable for seven days because the necessary attendant noninterruptible air compressor was out-of-service, was not returned to service nor was the unit placed in HOT SHUTDOWN within 12 hours and COLD SHUTDOWN in the following 24 hours.

This is a Severity Level III problem (Supplement I).
Civil Penalty - $100,000 (assessed equally between the violations).

Pursuant to the provisions of 10 CFR 2.201, Detroit Edison Company (Licensee) is hereby required to submit a written statement or explanation to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, within 30 days of the date of this Notice. This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each alleged violation: (1) admission or denial of the alleged violation; (2) the reasons for the violation, if admitted; (3) the corrective steps that have been taken and the results achieved; (4) the corrective steps that will be taken to avoid further violations; and (5) 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. Consideration may be given to extending the response time for good cause shown. Under the authority of Section 182 of the Act, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

Within the same time as provided for the response required under 10 CFR 2.201, the Licensee may pay the civil penalties by letter to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, with a check, draft, or money order payable to the Treasurer of the United States in the cumulative amount of the civil penalties proposed above, or may protest imposition of the civil penalties in whole or in part by a written answer addressed to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission. Should the Licensee fail to answer within the time specified, an Order imposing the civil penalties will be issued. Should the Licensee elect to file an answer in accordance with 10 CFR 2.205 protesting the civil penalties, in whole or in part, such answer should be clearly marked as an "Answer to a Notice of Violation" and may: (1) deny the violations listed in this Notice in whole or in part; (2) demonstrate extenuating circumstances; (3) show error in this Notice; or (4) show other reasons why the penalties should not be imposed. In addition to protesting the civil penalties, in whole or in part, such answer may request remission or mitigation of the penalties.

In requesting mitigation of the proposed penalties, the five factors addressed in Section V.B of 10 CFR, Part 2, Appendix C (1988), should be addressed. Any written answer in accordance with 10 CFR 2.205 should be set forth separately from the statement or explanation in reply pursuant to 10 CFR 2.201, but may incorporate parts of the 10 CFR 2.201 reply by specific reference (e.g., citing page and paragraph numbers) to avoid repetition. The attention of the licensee is directed to the other provisions of 10 CFR 2.205 regarding the procedure for imposing civil penalties.

Upon failure to pay any civil penalties due which subsequently have been determined in accordance with the applicable provision of 10 CFR 2.205, this matter may be referred to the Attorney General, and the penalties, unless compromised, remitted, or mitigated, may be collected by civil action pursuant to Section 234c of the Act, 42 U.S.C. 2282c.

The responses to the Director, Office of Enforcement, noted above (Reply to a Notice of Violation, letter with payment of civil penalties, and Answer to a Notice of Violation) should be addressed to: Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, with a copy to the Regional Administrator, Region III, U.S. Nuclear Regulatory Commission, 799 Roosevelt Road, Glen Ellyn, Illinois 60137 and a copy to the NRC Resident Inspector at Fermi.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Bert Davis

A. Bert Davis
Regional Administrator

Dated at Glen Ellyn, Illinois
this 16th day of June 1988

MAY 0 9 1988

Docket No. 50-341

EA 88-104

The Detroit Edison Company
ATTN: B. R. Sylvia,
Group Vice President
Nuclear Operations
6400 North Dixie Highway
Newport, MI 48166

Gentlemen:

This refers to the special safety inspection conducted by Messrs. W. G. Rogers and M. E. Parker of this office on October 18, 1987 through March 31, 1988, of activities at Fermi 2 authorized by Facility Operating License No. NPF-43 and to the discussion of our findings with Mr. W. Orser at the conclusion of the inspection. The inspection focused on the circumstances surrounding the inoperability of a reactor protection channel in excess of Technical Specification requirements and the design configuration of the primary containment radiation monitoring system.

The enclosed copy of our inspection report identifies areas examined during the inspection. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations, and interviews with personnel.

The apparent violation in Paragraph 3.d is being reviewed for potential enforcement action and was discussed as part of the Enforcement Conference held on April 28, 1988, between you and members of your staff and Dr. C. J. Paperiello and members of the NRC staff.

The apparent violations in Paragraph 2 are being reviewed for potential enforcement action in conjunction with the circumstances surrounding the failure to include required channel checks of the Reactor Protection System drywell high pressure instrumentation in any surveillance procedure (Licensee Event Report (LER) 87-048) and failure to perform the Division I Control Center Heating Ventilation and Air Conditioning chiller pump and valve operability surveillance by the critical completion date (LER 87-047). The inspection of these LERs will be documented in Inspection Report No. 50-341/88017(DRP). These problems, as a group, are similar to those that resulted in the Notice of Violation and Imposition of Civil Penalty of May 14, 1987, and appear to be indicative of a continued lack of overall understanding and appreciation of the Technical Specifications. Consequently,

it appears that the corrective actions described in your June 12, 1987, response to the May 14, 1987, Notice of Violation and Proposed Imposition of Civil Penalty have not been completely effective in preventing the recurrence of similar problems.

You will be notified by separate correspondence of our decision regarding enforcement action for these issues. No written response is required until you are notified of any proposed enforcement action.

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

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

Sincerely,

"Original Signed by E.G. Greenman"
Edward G. Greenman, Director
Division of Reactor Projects

Enclosure: Inspection Report
No. 50-341/87048(DRP)

cc w/enclosure:
Patricia Anthony, Licensing
P. A. Marquardt, Corporate
Legal Department
DCD/DCB (RIDS)
Licensing Fee Management Branch
Resident Inspector, RIII
Ronald Callen, Michigan
Public Service Commission
Harry H. Voight, Esq.
Michigan Department of
Public Health
Monroe County Office of
Civil Preparedness
J. Lieberman, OE
J. R. Goldberg, OGC
F. Miraglia, NRR

RIII	RIII	RIII	RIII	RIII	RIII
Pelke/jaw	Cooper	Knop	Grobe	Virgilio	Greenman
5/4/88	5/5/88	5/5/88	5/5/88	5/5/88	5/5/88

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
#*E. 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 (OSRC). 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 [illegible] 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-F401 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 during 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 1 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.

f. 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.

MAY 13 1988

Docket No. 50-341
EA 88-104

The Detroit Edison Company
ATTN: B. Ralph Sylvia,
Group Vice President
Nuclear Operations
6400 North Dixie Highway
Newport, MI 48166

Gentlemen:

This refers to the special safety inspection conducted by Mr. W. G. Rogers of this office on January 17, 1988 to April 28, 1988 of activities at Fermi 2 authorized by Facility Operating License No. NPF-43 and to the discussion of our findings with Mr. W. Orser at the conclusion of the inspection. The inspection examined the circumstances surrounding the failure of a Noninterruptible Control Air compressor on January 14, 1988.

The enclosed copy of our inspection report identifies areas examined during the inspection. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations, and interviews with personnel.

The apparent violation in Paragraph 5 is being reviewed for potential enforcement action and was discussed as part of the Enforcement Conference held on April 28, 1988, between you and members of your staff and Dr. C. J. Paperiello and members of the NRC staff.

You will be notified by separate correspondence of our decision regarding enforcement action for these issues. No written response is required until you are notified of any proposed enforcement action.

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

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

Sincerely,

8805260283

Edward G. Greenman, Director
Division of Reactor Projects

Enclosure: Inspection Report
No. 50-341/88014(DRP)

See Attached Distribution

RIII	RIII	RIII	RIII	RIII	RIII
Yes			Yes		
Pelke/ib	Cooper	Grobe	Knop	Virgo	Greenman
5-12-88		5/12/88	5-17-88	5/12/88	5/12

Distribution

cc w/enclosure:
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Legal Department
DCD/DCB (RIDS)
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Resident Inspector, RIII
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J. Lieberman, OE
L. J. Chandler, OGC
F. J. Miraglia, NRR

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-341/88014(DRP)

Docket No. 50-341 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: January 17, 1988 through April 28, 1988

Inspector: W. G. Rogers

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

RC Knop for

5-12-88
Date

Inspection Summary

Inspection on January 17, 1988 to April 28, 1988 (Report No. 50-341/88014(DRP))
Area Inspected: The circumstances and licensee actions surrounding the failure of a Noninterruptible Control Air compressor on January 14, 1988.
Results: One violation was identified (Paragraph 5). One open item was identified (Paragraph 5).

8805250276

DETAILS

1. Persons Contacted

a. Detroit Edison Company

F. Abramson, Operations Engineer
*P. Anthony, Licensing
#*S. Catola, Vice President, Nuclear Engineering and Services
*D. Gipson, Plant Manager
#*L. Goodman, Licensing Supervisor
*R. Lenart, General Director, Nuclear Engineering
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E. Wilds, Engineer
#L. Wooden, Supervisor, I&C Engineering
#B. Sylvia, Group Vice-President
#P. Marquardt, General Attorney
*W. Tucker, Superintendent, Operations

b. U. S. Nuclear Regulatory Commission

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#*W. Rogers, Senior Resident Inspector
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2. Background on Systems Involved

a. Compressed Air System - Updated Final Safety Analysis Report (UFSAR) Sections 9.3.1 and 7.6.1.17 discuss the compressed air system. The system description and operating modes provided below were derived from those UFSAR sections.

1. System Description

The air system is composed of two subsystems. The first is the station air subsystem. The second is the control air subsystem.

The station air subsystem (SAS) consists of three one-half-capacity 1225 scfm, two-stage nonlubricated reciprocating compressors equipped with inlet filter-silencers, and intercoolers and aftercoolers. Two 150 cubic foot capacity air receivers and the station air distribution piping, valves, and fittings complete the station air equipment. The station air subsystem is nonsafety-related.

The control air distribution subsystem is divided into two distinct parts: interruptible and noninterruptible.

The noninterruptible control air (NIAS) portion of the subsystem consists of two 100% capacity 100 scfm, single-stage nonlubricated reciprocating air compressors; two 100% capacity parallel strings of oil filters, air dryers, and afterfilters; two control air receivers; and associated piping, fittings, and valves. The noninterruptible control air is supplied through these two separate distribution systems (Division I and Division II) to the standby gas treatment system (SGTS), control center emergency filtration system (CCHVAC), main steam isolation valve leakage control system (MSIVLCS) and numerous other engineering safety features (ESF) systems. The noninterruptible control air portion of the control air subsystem is safety-related.

The interruptible control air (IAS) portion of the subsystem consists of a afterfilter; air dryer; air receiver; and associated piping, fittings, and valves. This portion of the subsystem is nonsafety-related.

2. Normal Operation

Normal Operation of the compressed air system is by air from the turbine building being drawn into one of the three SAS compressors. Air is compressed, cooled and discharged into the SAS receivers at 100 psig. The air is distributed to the SAS users through a header/riser system from the SAS receivers.

Air from the SAS subsystem is the source of air for the control air subsystem by connections to the SAS header.

Compressed air from the station air system is supplied through one of these connections to the Division I and II noninterruptible control air compressor discharge headers. The air then flows from each header through its divisional 100% capacity filter and dryer where it is cleaned of all particles of dirt and dried by a regenerative desiccant-type dryer. After leaving the filter/dryer, the noninterruptible control air flows to its point of use through its divisional noninterruptible control air distribution system.

Another SAS connection supplies the IAS. The air passes through the filter/dryer to the air receiver and then flows to its point of use through the IAS air distribution system.

3. Emergency Operation

On loss of offsite power, the SAS compressors lose electric power. The NIAS is isolated from the SAS by isolation valves sensing the low pressure in the SAS header. The control air compressors (CACs) are automatically started with power supplied from the emergency diesel generators. Enough receiver capacity is provided to supply 10 minutes of noninterruptible control air to allow sufficient time for the emergency diesel generators to supply power to the CACs and sufficient time for the CACs to pickup and carry the load. With normal offsite power available, the CACs start immediately on low SAS header pressure.

b. Main Steam Isolation Valve Leakage Control System (MSIVLCS) - UFSAR Section 6.2.6 discusses the MSIVLCS. The system description and operating modes were derived from that document.

1. The MSIVLCS consists of two redundant air-injection subsystems. Division I consists of the necessary piping and valving to permit injection of Division I control air into the above-seat drain on the four outboard MSIVs. This allows pressurization of the piping volume bounded by the four pairs of inboard and outboard MSIVs. Division II consists of the necessary piping and valving to permit injection of Division II control air into the main steam drain line upstream of the third MSIVs. This allows pressurization of the piping volume bounded by the four pairs of outboard MSIVs and a third set of motor-operated MSIVs.

2. During normal operation the system is not in use. The system is manually initiated approximately 20 minutes after the LOCA, when reactor pressure falls below 44 psig and the steam line pressure is less than 150 psig. Both divisions will pressurize their respective piping volumes to 2 to 6 psi above the reactor pressure. The system will continue to maintain this 2 to 6 psi difference above reactor pressure thus providing a positive sealing medium against the release of radioactivity from MSIV leakage.

c. Control Center Emergency Filtration System (CCHVAC) - UFSAR Section 9.4.1 discusses the CCHVAC. The system description and operating modes were derived from that document.

1. The CCHVAC consists of two 100% capacity air-conditioned supply units, an air distribution system, and an emergency filtration system. The control center is heated, cooled, and pressurized by a recirculating air system. CCHVAC processes control center

air and makeup air through charcoal filters. Air at 1800 cfm passes through two separate emergency air intakes to an emergency makeup air filter train. The filter train consists of a mist eliminator, two heaters, HEPA filter, charcoal filter and another HEPA filter. The emergency intake flow is then combined with 1200 cfm of control center recirculation airflow. This airflow is then processed through the recirculation air filter train. The emergency recirculation filter train consists of a prefilter, HEPA filter, charcoal filter and another HEPA filter. The air is drawn through these emergency filters by one of two redundant emergency recirculation air fans. Two redundant chilled water units are used to keep the air cool.

2. The system is automatically initiated by select loss of coolant signals. Upon initiation the proper damper/fan configuration is established to support this recirculation mode.

d. Standby Gas Treatment System (SGTS) - UFSAR Section 6.2.3 discusses the SGTS. The system description and operating modes were derived from that document.

1. The SGTS provides sufficient iodine removal capability following a loss of coolant accident. This capability is accomplished by pressurization of the seconoary containment preventing direct communication of contaminated air with the environment and filtration of contaminated air. The system consists of two separate and parallel 100% capacity trains. Ductwork allows for venting and purging of both the primary and the secondary containment atmospheres. In addition to the necessary ducts, controls, instrumentation, isolation valves, and protection systems each train consists of a moisture separator to remove entrained water droplets, a prefilter, electric heater, HEPA filter, deep-bed absorber unit, another HEPA filter, exhaust fan and a cooling air fan.

2. The system is automatically initiated on select loss of coolant signals. Upon initiation the proper fan/damper alignment is established to pressurize secondary containment.

3. Event Description

On January 14, 1988, the Division II NIAS control air compressor (CAC) failed during a periodic test. The licensee placed the CAC out of service, opened the crosstie valve to the Division I NIAS and initiated a work request.

On January 17, 1988, the Nuclear Shift Supervisor (NSS) instructed that Out of Specification Log (OSL) entry 88-060 be made on the out of service CAC. The OSL is the mechanism used by the licensee to determine the status of safety-related equipment required to be operable to satisfy Technical Specifications (TS) Limiting Conditions for Operations (LCO), thus ensuring that TS action statements are appropriately performed. The licensee utilizes a special type of OSL entry entitled a "tracking"

OSL entry. This "tracking" entry provides status on equipment that may invoke a TS action statement should other equipment be rendered inoperable. The January 17th OSL entry was a "tracking" type OSL. The entry was used to highlight the failed CAC increasing the management emphasis on its repair and return to service. The NSS did not consider the CAC required by TS and no action times were specified.

That same day the inspector noted the CAC to be out of service and questioned whether a TS system was affected by this equipment loss and a TS LCO action statement was in effect. The inspector informed the then Engineering Vice President of this concern.

On January 19, 1988, Deviation Event Report (DER) 88041 was written by engineering personnel on the impact of CAC failure as it relates to TS LCOs. The DER was dispositioned by engineering personnel on January 22. That disposition stated that since the three non-safety related station air compressors were operable, the Division I NIAS air compressor was operable and the cross-tie between NIAS Division I and NIAS Division II was open the Technical Specification associated systems would continue to receive the required control air necessary for safe plant shutdown. The DER disposition also stated that entrance into a 30 day LCO action statement was reasonable and prudent. The disposition did not state that a 30 day LCO action was required.

On January 24, 1988, OSL 88-060 was modified to identify a 30 day LCO action statement to be in affect from the date of CAC loss, January 14, 1988. The OSL entry did not specify the applicable TS which was requiring the 30 day LCO action and no actions were ever prescribed to be taken at the end of the 30 days. However, the licensee indicated that actions were being formulated to be taken at the end of the 30 days.

On February 3, 1988, the CAC was returned to service and OSL 88-060 was cleared. The unit was in Mode 1 during the whole period the CAC was out of service.

4. Inspector Followup

After questioning whether the CAC was a support system for TS systems the inspector began pursuing an answer to the question. Preliminary response to the question by licensed personnel was the CAC was not required.

The inspector reviewed the design specification for NIAS, drawings, design analysis of NIAS by Stone and Webster, design calculations for NIAS, Updated Final Safety Analysis Report (UFSAR) Section 9.3.1 and UFSAR Section 7.6.1.17. From reviewing these documents, all of which were active design basis documents, there were some discrepancies. These discrepancies were:

- The actual NIAS air users varied between the Stone and Webster analysis and the design specification.

- UFSAR Section 9.3.1.2 identified the reason for the crosstie between the IAS and NIAS to be for use during a NIAS Division II supply maintenance outage. The design specification Section 5 identified the reason as under circumstances where the station air system fails and the NIAS compressors start operation, plant personnel could remotely open the isolation valve provided the NIAS operation was not jeopardized.

After reviewing these design documents the inspector selected three air users that appeared to need the NIAS to perform their safety function. These systems were the main steam isolation valve leakage control system (MSIVLCS), standby gas treatment system (SGTS), and control center emergency filtration system (CCHVAC).

a. MSIVLCS Design Basis

The design and regulatory documents reviewed for MSIVLCS were:

- Design Specification 3071-530
- Regulatory Guide 1.96 Revision 1, Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants
- UFSAR Section 6.2.6
- UFSAR Appendix A, Conformance with Regulatory Guides
- TS 3.6.1.4
- TS 3.6.1.2
- Design Calculation 13067.13-P-B21-06-002.

The most salient sections of these documents were:

- Regulatory Guide 1.96, Revision 1, Section C.1. requires, in part, "The leakage control system and any necessary subsystems, including the source of any sealing fluid if a fluid seal type is used, should be designed in accordance with Seismic Category I and Quality Group B requirements..."
- Regulatory Guide 1.96, Revision 1, Section C.2. requires, in part, "The leakage control system (and any necessary subsystems) should be capable of performing its safety function, when necessary, considering effects resulting from a LOCA..."
- Regulatory Guide 1.96, Revision 1, Section C.3. states, "The leakage control system should be capable of performing its safety function following a LOCA and assumed single active failure (including failure of any one of the main steam isolation valves to close)."

- Regulatory Guide 1.96, Revision 1, Section C.5. states, "The leakage control system should be capable of performing its safety function following a loss of all offsite power coincident with a postulated design-basis LOCA."

- Regulatory Guide 1.96, Revision 1, Section C.6. states, "The leakage control system should be designed with sufficient capacity to control leakage from the main steam lines for as long as postulated accident conditions require containment integrity to be maintained."

- Appendix A of the UFSAR documents the licensee commitment to Regulatory Guide 1.96, Revision 1 with the exception of Section C.12 which is not applicable to the matters being addressed in this report.

- UFSAR Section 6.2.6.3 identifies single-failure criteria being accounted for in the design of the MSIVLCS by the seismic qualification of the main steam lines, the installation of a third MSIV in each main steam line, and the redundancy of active components and air sources.

- UFSAR Section 6.2.6.2 identifies MSIVLCS using two redundant air-injection systems: Divisions I and II. Division I MSIVLCS obtains air from the Division I control air system. Division II MSIVLCS obtains air from the Division II control air system. These divisions would be manually activated 20 minutes after a postulated LOCA.

- TS 3.6.1.4 requires two independent MSIV leakage control system subsystems to be operable with specific remedial action to be taken when a subsystem is inoperable.

- UFSAR Section 6.2.6.3 states in part "Gross MSIV leakage would not result in a degradation of the positive-seal MSIVLCS. Gross leakages on the order of 1000 scfh are well within the capacity of the Category I control air system, which is the source of air for the MSIVLCS. The maximum injection rate of air through the MSIVLCS to the main steam piping is limited to 50 scfm..."

- TS 3.6.1.2.c establishes the maximum acceptable leakages rates to be less than or equal to 100 scfh for all four main steam lines when tested at 25 psig.

- Design Specification 3071-530 Sections 7.3.1 and 7.3.2 identify MSIVLCS to be subject to a design bases accident of 180 days.

- The design calculation 13067.13-P-B21-06-002 determined that 126 minutes would be needed to pressurize the Division II piping and 20 minutes for the Division I piping at a fill rate of 25 scfm.

Upon completion of this design review the inspector determined that each division of MSIVLCS must be capable of performing its safety function even with a leak of 100 scfh present for 180 days following the LOCA. With a leak of this magnitude in either division of the MSIVLCS both NIAS air receiver tanks (341 cubic feet volume each) would be depleted long before the end of the first day following the LOCA. To maintain MSIVLCS capable of performing its safety function requires a CAC to be operable. To meet the requirement of being able to perform its safety function with a single active failure present requires two divisions of MSIVLCS with two air sources, i.e., two CACs. Also the air usage for the MSIVLCS stated in the Stone and Webster analysis was not consistent with the steam line initial pressurization usage used in design calculation 13067.13-P-B21-06-002.

b) CCHVAC and SGTS Design Basis

The design basis and regulatory documents reviewed were:

- Applicable Functional Operating Sketches
- Applicable P&IDs
- Detailed component drawings
- Regulatory Guide 1.52, Revision 2; Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants
- UFSAR Appendix A
- UFSAR Section 6.2.3
- UFSAR Section 9.4.1
- TS 3.6.5.3
- TS 3.7.2

The most salient sections of these documents were:

- Regulatory Guide 1.52, Rev. 2, Section C.2.a states in part "ESF atmosphere cleanup systems designed and installed for the purpose of mitigating accident doses should be redundant."
- Regulatory Guide 1.52, Rev. 2, Section C.2.h requires all instrumentation and equipment controls be designed to IEEE Standard 279 and Section 4.7 of IEEE Standard 279 requires protection against single failure.

- UFSAR Appendix A documents the licensee's commitment to Regulatory Guide 1.52 stating that the CCHVAC active components (fans, dampers, controls, etc.) are redundant and meet IEEE 279.

- UFSAR Section 9.4.1.1.d states for the CCHVAC, "Redundant components are powered by their corresponding redundant Division I and Division II engineered safety feature buses."

- UFSAR Section 9.4.1.1.e states for the CCHVAC, "The system is designed to accomplish its design objectives assuming a single active component failure."

- TS 3.7.2 requires the control room emergency filtration system to be operable and requires specific remedial action be taken when a required redundant component is inoperable.

- Dampers T41-F69B, 38, 31B, 61B, 56B, 56A, 54, 40B, 34B, 34D, 47 48, 66 and 67 do not fail to their safety function position for CCHVAC recirculation upon loss of air pressure. There are no accumulators or reservoirs for these dampers.

- UFSAR Section 6.2.3.2 states, in part, "The SGTS is a 100%-redundant ESF system..." and "...consists of two separate and parallel 100% capacity trains."

- UFSAR Section 6.2.3.3.1 states, in part, "All power and control circuits meet the requirements of IEEE 279. Redundant active components are provided where necessary to ensure that a single failure does not impair or prevent system operation."

- TS 3.6.5.3 requires two independent standby gas treatment subsystems to be operable.

- Dampers T46-F01A, 01B, 02B, 02A, 03A, 03B, 04A, 04B, 05A, 05B, 07A, 07B, 08A ,08B, 406, 407, 408 and 409 do not fail to their safety function position upon loss of air pressure. There are no accumulators or reservoirs for these dampers.

It was apparent from these document reviews that air pressure is required to place necessary SGTS and CCHVAC dampers in the safety function position, without which the systems are inoperable. The air pressure for these systems must be from NIAS. Since these systems are required to be redundant, two divisions of NIAS including CACs are necessary to support these systems. Without CACs the MSIVLCS usage depletes the air receivers and the SGTS/CCHVAC dampers fail closed rendering the CCHVAC and SGTS incapable of passing air flow.

The inspector presented the review of the NIAS/SGTS/CCHVAC/MSIVLCS information to the licensee engineering staff who agreed with the inspector's conclusion that CACs are necessary to support MSIVLCS, SGTS and CCHVAC system performance. However, the licensee contended that no

LCO action statement is invoked when a CAC is out of service. The rationale for this statement was based on the engineering staff's interpretation of a statement in Section 9.2.1.2 of the UFSAR. The statement is "There is a normally closed intertie between the Divisions I and II noninterruptible control air systems." During a maintenance outage of the supply to one of these divisions, the intertie is opened so that the division having the outage can be supplied by the other division. The licensee believed that the statement in the UFSAR gave authorization to remove a CAC from service for maintenance for an indeterminate period of time and this condition was a part of the original design basis of the system. Additionally, the engineering staff considered one CAC to have adequate capacity for both divisions of air users.

The inspector informed the licensee that this philosophy was not consistent with the regulatory requirements and began pursuing whether this philosophy had manifested itself in other areas of the licensee organization.

a) Current Training Review - The inspector reviewed the current operator, licensed and non-licensed, training lesson guides on CCHVAC and the compressed air system and ascertained:

   - The CCHVAC training guides state that control air supplies air operators for dampers under the "Interrelationships with Other Plant Systems" section.

   - The compressed air training guide states in the Technical Specifications section "The station and control air system is not specifically mentioned in Technical Specifications, however, many systems which are mentioned in Tech. Specs. do require control air system operability to be considered operable."

   - The compressed air training guide does not explicitly state those systems rendered inoperable by loss of control air.

b) DER Review - The inspector reviewed past deviation reports (DERs) associated with the control air system and ascertained:

   - DER 85-0667 discussed a situation where the Division I/II intertie and the NIAS/IAS intertie were used simultaneously. The DER disposition related to the UFSAR sections on the interties and concluded that this situation was not a design deficiency.

   - DER 87-0322 discussed a postulated feedwater line break which would render the three nonsafety-related station air compressors out of service by the high energy fluid and the two NIAS compressors out of service through flooding of their respective rooms. The licensee analyzed this condition assuming the plant was in Mode 1 at initiation of the event. The conclusion was that safe shutdown could be achieved. However, the conclusion was based on manual repositioning of CCHVAC dampers and that the

SGTS was not required. The DER further states "Loss of control air and loss of feedwater are part of UFSAR Chapter 15 analysis. In addition, FW line break with subsequent loss of control air effect evaluation was performed. The evaluation shows that for the FW line break with loss of control air, safe shutdown can be achieved."

- Engineering staff personnel were a party to the disposition of DER 87-0322 and DER 85-0667.

With regard to DER 85-0667 the inspector pursued the use of both interties simultaneously with the engineering staff. After questioning the engineering staff, they stated that this configuration was outside the design basis. The inspector reviewed the procedures to determine whether this was expressly forbidden in the procedures. The procedures did not forbid such actions.

With regard to DER 87-0322 the two assumptions are invalid in that manual repositioning of the CCHVAC dampers from outside the control room is outside the design basis for CCHVAC and the SGTS would receive an automatic initiation signal (including damper positioning) at a reactor vessel level 2 signal. The conclusion is not correct given the invalidation of the two assumptions in the analysis.

c) Operator Interviews - The inspector discussed the NIAS with a large number of senior reactor operators. Those interview results were:

- The licensed individuals stated that they had not been trained or directed to consider the NIAS as a Technical Specification support system.

- Loss of a CAC did not invoke any LCO action statements.

- Use of the interties was acceptable.

d) Independent Safety Evaluation Group (ISEG) Reviews - In 1987 as a response to a Notice of Violation, the licensee committed to have ISEG review the testing of select safety related systems. One of these systems was compressed air. The inspector reviewed that report and concluded that the deficiencies identified in this inspection report were outside the scope of the ISEG review.

e) Procedure Content - The inspector reviewed system operating Procedure 23.129, Station and Control Air System. Section 1.1 states, in part, "The NIAS is provided to be interconnected, should one divisional supply be lost."

It was apparent from these five reviews that the engineering interpretation regarding NIAS had been adopted by the production organization.

Finally, the inspector requested to review the calculations supporting single CAC operation feeding both user divisions of NIAS. The licensee was unable to provide any calculations of this nature. The inspector requested this calculation be performed to determine if by opening the crosstie the MSIVLCS, SGTS and CCHVAC systems would perform their safety functions or the control air demand would be greater than the safety related air sources (two air receivers and one CAC) could provide. The calculation was performed and completed in late April as design calculation 4931. The calculation supported the licensee's position.

5. Conclusion

Upon completion of this inspection the inspector concluded that:

a. 10 CFR 50.36(c)(2) states that Technical Specification Limiting Conditions for Operation are "the lowest functional capabilities or performance levels of equipment required for safe operation of the facility." Technical Specifications Limiting Conditions for Operation 3.6.5.3, 3.7.2 and 3.6.1.4 require two redundant operable subsystems for standby gas treatment, control center emergency filtration (active components only) and main steam isolation valve leakage control. The intent behind the necessity for two subsystems is to assure that a single active failure does not render the systems incapable of performing their safety functions.

10 CFR 50.36(c)(2) further states, "When a LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specification until the condition can be met." If these systems are not capable of performing their safety function with a single failure present then their respective LCO action statements must be invoked since the LCO requiring two redundant subsystems is not met.

The NIAS is a TS support system as defined be TS 1.25 which states, "A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified functions and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s)."

During the time the CAC was out of service the licensee did not enter into the LCO action statements of Technical Specification 3.6.5.3, 3.7.2 or 3.6.1.4. With the Division II CAC out of service these systems will not perform their safety function if a single failure were to occur to the Division I CAC, the EDG which supplies power to the Division I CAC or an abnormal air demand on the NIAS system. The CCHVAC and SBGT LCO action statements allow operation for up the 7 days before unit shutdown must commence. The MSIVLCS LCO action statement allows operation for up to 30 days before unit shutdown must commence.

On January 21, 1988, when the 7 day LCO expired the licensee failed to place the unit in Hot Shutdown by 2215 on January 21, 1988, and Cold Shutdown by 2215 on January 22, 1988. This is considered a violation (50-341/88014-01(DRP)) of Technical Specifications 3.6.5.3 and 3.7.2.

The root cause of this violation was the inadequate understanding of the underlying design bases for the NIAS as it supports the operability of Technical Specification systems by the engineering organization. It appears that the engineering organization has not provided the appropriate direction to the rest of the Fermi 2 organization. Therefore, the training personnel have not told the operators of the direct operability tie between NIAS and SBGT/CCHVAC/MSIVLCS and the procedure writers have not provided that guidance in the operating procedures.

b. The design basis documents associated with the NIAS have not been kept current and provide inconsistent information on the NIAS. Reconciliation of the documents is considered an open item (50-341/88014-02(DRP)).

The root cause of this matter was a lack of coordination between the engineers associated with the NIAS and the engineers associated with the NIAS air users. Each thought the other was providing the necessary information in their respective categories.

c. Guidance should be provided to the operating shifts forbidding operation of both compressed air interties simultaneously.

6. Exit Interview (30703)

The inspectors met with licensee representatives (denoted in Paragraph 1) 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. However, the licensee management supported the licensee engineering staff in that management considered that no violation of TS LCOs had occurred and the DER 88041 disposition was proper.

7. Enforcement Conference

On April 28, 1988, an enforcement conference was held on the NIAS CAC situation and design deficiencies of the primary containment monitoring system (PCMS) discussed in Inspection Report 50-341/87048(DRP).

With respect to the NIAS portion of the conference the licensee restated their position that a LCO action statement had not been involved with a NIAS CAC out of service. The inspector restated his conclusions as to why a LCO was applicable.

New information provided at the conference was:

- The licensee is considering a TS change explicit for NIAS .
- The licensee performed a probabilistic risk assessment of the ramifications of having the intertie open and closed with one CAC in service.
- PCMS operability is also affected in the same manner as SGTS/CCHVAC/MSIVLCS upon loss of control air.