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II Major CECo Submittals Referenced in Fire Protection Safety Evaluation Reports (see Vol. 2 of FPPDP)

III General Fire Protection Correspondence (see Vol. 2 of FPPDP)

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III Fire Rated Barrier Design Support Data (see Vol. 6 of FPPDP)

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V Supplement One - Hydraulic Verification Study (see Vol. 11 of FPPDP)

VI Supplement Two - Hydraulic Verification Study (see Vol. 11 of FPPDP)

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VIII Supplement Four - Hydraulic Verification Study (see Vol. 11 of FPPDP)

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DRESDEN 2 & 3

FIRE PROTECTION REPORT

Volume Index

Tab I

DRESDEN 2&3

FIRE PROTECTION PROGRAM DOCUMENTATION PACKAGE

Fire Protection Technical Specifications and License Condition

Fire protection technical specifications have been removed per Generic Letter 86-10, as discussed in the following letters which are included in this section:

1 February 22, 1989 CECo letter from J. A. Silady (NLA) to T. E. Murley (NRC) transmitting a proposed amendment to replace the Fire Protection Technical Specifications with a standard license condition and appropriate administrative procedures.

2 June 30, 1989 NRC letter from B. L. Siegel to T. J. Kovach (CECo) issuing Technical Specification amendments to replace the existing license conditions on fire protection with the standard condition noted in Generic Letter 86-10.

3 August 9, 1989 NRC letter from B. L. Siegel to T. J. Kovach (CECo) issuing corrected index pages for the Technical Specifications amendments provided by letter dated June 30, 1989.

Tab 1

Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690 - 0767

February 22, 1989

Dr. Thomas E. Murley, Director
Office of Nuclear Reactor Regulation
ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Dresden Nuclear Power Station Units 2 and 3
Proposed Amendment to the Fire Protection
License Condition and Technical Specifications
NRC Docket Nos. 50-237 and 50-249

Reference: Generic Letters 86-10 and 88-12 dated April 24, 1986 and August 2, 1988, respectively.

Dear Dr. Murley:

Pursuant to 10 CFR 50.90, Commonwealth Edison proposes to amend Provisional Operating License DPR-19 and Facility Operating License DPR-25 for Dresden Nuclear Power Station and their respective Appendix A Technical Specifications. The proposed amendment revises the Units 2 and 3 Licenses and Technical Specifications in response to the referenced Generic Letters and as part of the Dresden Improvement Program's Technical Specification Action Plan.

The referenced Generic Letters suggested replacement of Fire Protection Technical Specifications with a standard license condition and appropriate administrative procedures, after updating the FSAR to reflect the approved fire protection program. Similar amendments have been previously approved, such as Amendment 10 to the Byron Station Operating Licenses (NPF-37 and NPF-66) issued September 9, 1987. The changes are summarized in Attachment 1 and further described in Attachment 3. The affected pages of the Licenses and Technical Specifications are contained in Attachment 2.

The proposed changes have been reviewed and approved by both On-Site and Off-Site Review in accordance with Commonwealth Edison procedures. We have reviewed these proposed amendments in accordance with 10 CFR 50.92(c) and determined that no significant hazards consideration exists. This evaluation is documented in Attachment 4.

Enclosed as Attachment 5 are the proposed Dresden Administrative Technical Requirements (DATRs) for Fire Protection. They are submitted as supporting information for this amendment but have not yet been approved for implementation by On-Site and Off-Site Review. Since some DATR provisions are different from existing Technical Specifications, they cannot be fully

implemented until the amendment has been issued. Although some minor changes which should not affect their technical content may be required prior to final on-site and off-site review and approval, CECo believes the enclosed preliminary version to be technically adequate to support the Staff's review of the proposed amendment.

Commonwealth Edison is notifying the State of Illinois of our application for this amendment by transmitting a copy of this letter and its attachments to the designated State Official.

Please direct any questions you may have regarding this matter to this office.

Very truly yours,

J A Silady

J. A. Silady
Nuclear Licensing Administrator

lm

Attachments 1: Summary of Changes
2: Proposed Changes to Appendix A Technical Specifications for Dresden Units 2 and 3
3: Description and Bases for Amendment Request
4: Significant Hazards Evaluation
5: Proposed Dresden Administrative Technical Requirements for Fire Protection

cc: A.B. Davis - Regional Administrator, RIII
D.R. Muller - Project Director, NRR
S.G. DuPont - NRC Senior Resident Inspector, Dresden
B.L. Siegel - Project Manager, NRR
D.R. Hoffman - Excel Services
Office of Nuclear Facility Services - IDNS

SUBSCRIBED AND SWORN to
before me this 22nd day
of February, 1989

[illegible] Mayo
Notary Public

5554K

ATTACHMENT I

SUMMARY OF CHANGES

The following changes have been identified for Dresden Units 2 and 3:

(1) Page 4 of license (DPR-19 and 25)

Change the license condition Section 3.H for Unit 2 and Section 3.G for Unit 3 to the standard fire protection license condition identified in Generic Letter 86-10.

(2) Pages 3/4.12-1 through 3/4.12-21 (DPR-19 and 25)

Delete all sections of the fire protection Technical Specifications.

(3) Page 6-1 (DPR 19 and 25)

Delete manning requirements for fire brigade.

(4) Page 6-7 (DPR-19 and 25)

Add new Section 6.1.G.1.a.11 which states that the responsibilities of the Off-Site Review and Investigative Function shall include the review of changes to the Fire Protection Program and implementing procedures.

(5) Page 6-13 (DPR-19 and 25)

Add new Section 6.1.G.2.a.13 which states that the responsibilities of the On-Site Review and Investigative Function shall include the review of changes to the Fire Protection Program and implementing procedures.

3055a

ATTACHMENT 2

PROPOSED CHANGES TO LICENSE AND

TECHNICAL SPECIFICATIONS

UNIT 2 PAGES AFFECTED (DPR-19)	UNIT 3 PAGES AFFECTED (DPR-25)
License Page 4	License Page 4
3/4.12-1	3/4.12-1
3/4.12-2	3/4.12-2
3/4.12-3	3/4.12-3
3/4.12-4	3/4.12-4
3/4.12-5	3/4.12-5
3/4.12-6	3/4.12-6
3/4.12-7	3/4.12-7
3/4.12-8	3/4.12-8
3/4.12-9	3/4.12-9
3/4.12-10	3/4.12-10
3/4.12-11	3/4.12-11
3/4.12-12	3/4.12-12
3/4.12-13	3/4.12-13
3/4.12-14	3/4.12-14
B 3/4.12-15	B 3/4.12-15
B 3/4.12-16	B 3/4.12-16
B 3/4.12-17	B 3/4.12-17
B 3/4.12-18	B 3/4.12-18
B 3/4.12-19	B 3/4.12-19
B 3/4.12-20	B 3/4.12-20
B 3/4.12-21	B 3/4.12-21
6-1	6-1
6-7	6-7
6-13	6-13

Tab 2

Original File
cc: I. Peterson / K. Bauer
M. Dillon
M. Strait / B. Bart

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

June 30, 1989

Revision 8
April 1992

Docket Nos.: 50-237
and 50-249

Mr. Thomas J. Kovach
Nuclear Licensing Manager
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Kovach:

SUBJECT: REPLACE FIRE PROTECTION LICENSE CONDITION AND REMOVAL OF FIRE PROTECTION TECHNICAL SPECIFICATIONS AS PER GENERIC LETTERS 86-10 AND 88-12 (TAC NOS. 71256 AND 71257)

Re: Dresden Nuclear Power Station, Unit Nos. 2 and 3

The Commission has issued the enclosed Amendment No. 106 to Provisional Operating License No. DPR-19 for Dresden Unit 2 and Amendment No. 101 to Facility Operating License No. DPR-25 for Dresden Unit 3. These amendments are in response to your application dated February 22, 1989.

The aforementioned amendments replace the existing license conditions on fire protection with the standard condition noted in Generic Letter 86-10 and remove requirements for fire detection systems, fire suppression systems, fire barriers and fire brigade staffing requirements as per guidance contained in Generic Letter 86-10 and 88-12.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notices.

RECEIVED
JUL 10 1989

Sincerely,

Byron L. Siegel

Byron L. Siegel, Project Manager
Project Directorate III-2
Division of Reactor Projects III,
IV, V, and Special Projects

Enclosures:
1. Amendment No. 106 to
License No. DPR-19
2. Amendment No. 101 to
License No. DPR-25
3. Safety Evaluation

cc w/enclosures:
See next page

Mr. Thomas J. Kovach
Commonwealth Edison Company

Dresden Nuclear Power Station
Units 2 and 3

cc:

Michael I. Miller, Esq.
Sidley and Austin
One First National Plaza
Chicago, Illinois 60603

~~Mr. J. Eenigenburg~~
Plant Superintendent
Dresden Nuclear Power Station
Rural Route #1
Morris, Illinois 60450

U. S. Nuclear Regulatory Commission
Resident Inspectors Office
Dresden Station
Rural Route #1
Morris, Illinois 60450

Chairman
Board of Supervisors of
Grundy County
Grundy County Courthouse
Morris, Illinois 60450

Regional Administrator
Nuclear Regulatory Commission, Region III
799 Roosevelt Road, Bldg. #4
Glen Ellyn, Illinois 60137

Mr. Michael E. Parker, Chief
Division of Engineering
Illinois Department of Nuclear Safety
1035 Outer Park Drive, 5th Floor
Springfield, Illinois 62704

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION, UNIT NO. 2

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 106
License No. DPR-19

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated February 22, 1989 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraphs 3.B. and 3.H. of Provisional Operating License No. DPR-19 are hereby amended to read as follows:

   B. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 106 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   H. Commonwealth Edison Company shall implement and maintain in effect all provisions of the approved fire protections program as described in the Updated Final Safety Analysis Report for the facility and as approved in the SERs dated March 22, 1978 with supplements dated December 2, 1980 and February 12, 1981; January 19, 1983; July 17, 1987; September 28, 1987; and January 5, 1989, subject to the following provisions:

      The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

3. This license amendment is effective as of the date of its issuance to be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Paul C. Shemanski

Paul C. Shemanski, Acting Director
Project Directorate III-2
Division of Reactor Projects - III, IV, V and Special Projects

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 30, 1989

Revision 8
April 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 106

PROVISIONAL OPERATING LICENSE DPR-19

DOCKET NO. 50-237

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
iv	iv
3/4.12-1	
3/4.12-2	
3/4.12-3	
3/4.12-4	
3/4.12-5	
3/4.12-6	
3/4.12-7	
3/4.12-8	
3/4.12-9	
3/4.12-10	
3/4.12-11	
3/4.12-12	
3/4.12-13	
3/4.12-14	
B 3/4.12-15	
B 3/4.12-16	
B 3/4.12-17	
B 3/4.12-18	
B 3/4.12-19	
B 3/4.12-20	
B 3/4.12-21	
6-1	6-1
6-7	6-5
6-13	6-11

(Table of Contents, Cont'd.)

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization, Review, Investigation and Audit

A. Onsite and offsite organizations shall be established for the unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

1. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through the intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of department responsibilities and relationships, and job descriptions for key personnel positions, or in the equivalent forms of documentation. The requirements shall be documented in the Quality Assurance Manual or the Management Plan for Nuclear Operations, Section 3 Organizational Authority, Activity; Section 6 Interdepartmental Relationships.

2. The Station Manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of plant.

3. The Senior Vice President-Nuclear Operations shall have the corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

4. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operational pressures.

B. DELETED

C. The shift manning for the station shall be as shown in Table 6.1.1. The Operating Assistant Superintendent, Operating Engineers, Shift Engineers, and Shift Foremen shall have a Senior Operating License. The Fuel Handling Foreman shall have a limited Senior Operating License. The Vice President BWR Operations on the corporate level has responsibility for the Fire Protection Program. An Operating Engineer at the station will be responsible for implementation of the Fire Protection Program.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

(5) Noncompliance with NRC requirements, or of internal procedures or instructions having nuclear safety significance.

(6) Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety as referred to it by the On-site Review and Investigative Function.

(7) Reportable events under 10 CFR 50.73.

(8) All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems or components.

(9) Review and report findings and recommendations regarding all changes to the Generating Stations Emergency Plan prior to implementation of such changes.

(10) Review and report findings and recommendations regarding all items referred by the Technical Staff Supervisor, Station Manager, Vice President BWR Operations and AVP Quality Programs and Assessment.

(11) Review changes to the Fire Protection Program and implementing procedures.

b. Station Audit Function

The Station Audit Function shall be the responsibility of the AVP Quality Programs and Assessment independent of BWR Operations. Such responsibility is delegated to the Nuclear Quality Programs Manager.

Either of the above, or designated Corporate Staff or Supervisor approved by AVP Quality Programs and Assessment shall approve the audit agenda and checklists, the findings and the report of each audit. Audits shall be performed in accordance with the Company Quality Assurance Program and Procedures. Audits shall be performed to assure that safety-related functions are covered within the period designated below:

(1) Audit of the Conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per year.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

(7) Performance of special reviews and investigations and reports thereon as requested by the Superintendent of the Off-site Review and Investigative Function.

(8) Review of the Station Security Plan and shall submit recommended changes to the Director of Corporate Security and the AVP Quality Programs and Assessment in lieu of distribution in accordance with 6.1.G.2.c.(1).

(9) Review of the Emergency Plan and station implementing procedures and identification of recommended changes.

(10) Review of reportable events and actions taken to prevent recurrence.

(11) Review of any unplanned on-site release of radioactive material to the environs including the preparation and forwarding of reports covering evaluation recommendations and disposition of the corrective action to prevent recurrence to the Vice President BWR Operations and to the Superintendent of the Off-site Review and Investigative Function.

(12) Review of changes to the PCP and ODCM and major changes to the radwaste treatment systems.

(13) Review changes to the Fire Protection Program and implementing procedures.

b. Authority

The Technical Staff Supervisor is responsible to the Station Manager (or designee) and shall make recommendations in a timely manner in all areas of review, investigation, and quality control phases of plant maintenance, operation and administrative procedures relating to facility operations. The Technical Staff Supervisor shall have the authority to request the action necessary to ensure compliance with rules, regulations, and procedures when in his opinion such action is necessary. The Station Manager (or designee) shall follow such recommendations or select a course of action that is more conservative regarding safe operation of the facility. All such disagreements shall be reported immediately to the Vice President BWR Operations and the Superintendent of the Off-site Review and Investigative Function.

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

DRESDEN NUCLEAR POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 101
License No. DPR-25

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated February 22, 1989 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter 1;

   B. The facility will operate in conformity with the application, the provisions of the Act and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraphs 3.B. and 3.G. of Facility Operating License No. DPR-25 are hereby amended to read as follows:

    B. <u>Technical Specifications</u>

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 101 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

    G. Commonwealth Edison Company shall implement and maintain in effect all provisions of the approved fire protections program as described in the Updated Final Safety Analysis Report for the facility and as approved in the SERs dated March 22, 1978 with supplements dated December 2, 1980 and February 12, 1981; January 19, 1983; July 17, 1987; September 28, 1987; and January 5, 1989, subject to the following provisions:

    The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

3. This license amendment is effective as of the date of its issuance to be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Paul C. Shemanski

Paul C. Shemanski, Acting Director
Project Directorate III-2
Division of Reactor Projects - III, IV, V and Special Projects

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 30, 1989

ATTACHMENT TO LICENSE AMENDMENT NO.101

FACILITY OPERATING LICENSE DPR-25

DOCKET NO. 50-249

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
iv	iv
3/4.12-1	
3/4.12-2	
3/4.12-3	
3/4.12-4	
3/4.12-5	
3/4.12-6	
3/4.12-7	
3/4.12-8	
3/4.12-9	
3/4.12-10	
3/4.12-11	
3/4.12-12	
3/4.12-13	
3/4.12-14	
B 3/4.12-15	
B 3/4.12-16	
B 3/4.12-17	
B 3/4.12-18	
B 3/4.12-19	
B 3/4.12-20	
B 3/4.12-21	
6-1	6-1
6-7	6-5
6-13	6-11

(Table of Contents, Cont'd.)

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization, Review, Investigation and Audit

A. Onsite and offsite organizations shall be established for the unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

1. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through the intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form or organization charts, functional descriptions of department responsibilities and relationships, and job descriptions for key personnel positions, or in the equivalent forms of documentation. The requirements shall be documented in the Quality Assurance Manual or the Management Plan for Nuclear Operations, Section 3 Organizational Authority, Activity; Section 6 Interdepartmental Relationships.

2. The Station Manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of plant.

3. The Senior Vice President-Nuclear Operations shall have the corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

4. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operational pressures.

B. DELETED

C. The shift manning for the station shall be as shown in Table 6.1.1. The Operating Assistant Superintendent, Operating Engineers, Shift Engineers, and Shift Foremen shall have a Senior Operating License. The Fuel Handling Foreman shall have a limited Senior Operating License. The Vice President BWR Operations on the corporate level has responsibility for the Fire Protection Program. An Operating Engineer at the station will be responsible for implementation of the Fire Protection Program.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

(4) Proposed changes in Technical Specifications or NRC operating licenses.

(5) Noncompliance with NRC requirements, or of internal procedures or instructions having nuclear safety significance.

(6) Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety as referred to it by the Onsite Review and Investigative Function.

(7) Reportable Events reported under 10 CFR 50.73.

(8) All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems or components.

(9) Review and report findings and recommendations regarding all changes to the Generating Stations Emergency Plan prior to implementation of such changes.

(10) Review and report findings and recommendations regarding all items referred by the Technical Staff Supervisor, Station Manager, Vice President BWR Operations and AVP Quality Program and Assessment.

(11) Review changes to the Fire Protection Program and implementing procedures.

b. Station Audit Function

The Station Audit Function shall be the responsibility of the AVP Quality Programs and Assessment independent of the Production Department. Such responsibility is delegated to the Nuclear Quality Programs Manager.

Either of the above, or designated Corporate Staff or Supervision approved by AVP Quality Programs and Assessment shall approve the audit agenda and checklists, the findings and the report of each audit. Audits shall be performed in accordance with the Company Quality Assurance Program and Procedures. Audits shall be performed to assure that safety-related functions are covered within the period designated below.

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

(6) Review of facility operations to detect potential safety hazards.

(7) Performance of special reviews and investigations and reports thereon as requested by the Superintendent of the Off-site Review and Investigative Function.

(8) Review the Station Security Plan and shall submit recommended changes to the Director of Corporate Security and the AVP Quality Programs and Assessment in lieu of distribution in accordance with 6.1.G.2.c(1).

(9) Review the Emergency Plan and station implementing procedures and identification of recommended changes.

(10) Review of reportable events and actions taken to prevent recurrence.

(11) Review of any unplanned on-site release of radioactive material to the environs including the preparation and fowarding of reports covering evaluation recommendations and disposition of the corrective action to prevent recurrence to the Vice President BWR Operations and to the Superintendent of the Off-site Review and Investigative Function.

(12) Review of changes to the PCP and ODCM and major changes to the radwaste treatment systems.

(13) Review changes to the Fire Protection Program and implementing procedures.

b. Authority

The Technical Staff Supervisor is responsible to the Station Manager (or designee) and shall make recommendations in a timely manner in all areas of review, investigation, and quality control phases of plant maintenance, operation and administrative procedures relating to facility operations. The Technical Staff Supervisor shall have the authority to request the action necessary to ensure compliance with rules, regulations, and procedures when in his opinion such action is necessary. The Station Manager (or designee) shall follow such recommendations or select a course of action that is more conservative regarding safe operation of the facility. All such disagreements shall be reported immediately to the Vice President BWR Operations and the Superintendent of the Off-site Review and Investigative Function.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 106 TO PROVISIONAL OPERATING LICENSE NO. DPR-19

AND AMENDMENT NO. 101 TO FACILITY OPERATING LICENSE NO. DPR-25

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNIT NOS. 2 AND 3

DOCKET NOS. 50-237 AND 50-249

1.0 INTRODUCTION

By letter dated February 22, 1989, Commonwealth Edison Company (the licensee) proposed that the existing license conditions on fire protection be replaced with the standard condition noted in Generic Letter 86-10 and also proposed changes to the Appendix A Technical Specifications (TS) for Dresden Units 2 and 3. The proposed changes would remove requirements for fire detection systems, fire suppression systems, fire barriers, and fire brigade staffing requirements as recommended by Generic Letter 86-10. The proposed changes would also modify the administrative control requirements of the TS to add requirements for the Fire Protection Program that are similar to requirements for other programs implemented by license condition. Guidance on these proposed changes to TS was provided to all power reactor licensee and applicants by Generic Letter 88-12, dated August 2, 1988.

2.0 BACKGROUND

Following the fire at the Browns Ferry Nuclear Power Plant on March 22, 1975, the Commission undertook a number of actions to ensure that improvements were implemented in the Fire Protection Programs for all power reactor facilities. Because of the extensive modification of Fire Protection Programs and the number of open issues resulting from staff evaluations, a number of revisions and alterations occurred in these programs over the years. Consequently, licensees were requested by Generic Letter 86-10 to incorporate the final NRC approved Fire Protection Program in their Final Safety Analysis Reports (FSARs). In this manner, the Fire Protection Program -- including the systems, the administrative and technical controls, the organization, and other plant features associated with fire protection -- would have a status consistent with that of other plant features described in the FSAR. In addition, the Commission concluded that a standard license condition, requiring compliance with the provisions of the Fire Protection Program as described in the FSAR, should be used to ensure uniform enforcement of fire protection requirements. Finally, the Commission stated that with the requested actions, licensees may request an amendment to delete the fire protection TS that would now be unnecessary.

The licensees for the Callaway and Wolf Creek plants submitted lead-plant proposals to remove fire protection requirements from their TS. This action was an industry effort to obtain NRC guidance on an acceptable format for license amendment requests to remove fire protection requirements from TS.

Additionally, in the licensing review of new plants, the staff has approved applicant requests to remove fire protection requirements from TS issued with the operating license. Thus, on the basis of the lead-plant proposals and the staff's experience with TS for new licenses, Generic Letter 88-12 was issued to provide guidance on removing fire protection requirements from TS.

3.0 EVALUATION

Generic Letter 86-10 recommended the removal of fire protection requirements from the TS. Although a comprehensive Fire Protection Program is essential to plant safety, the basis for this recommendation is that many details of this program that are currently addressed in TS can be modified without affecting nuclear safety. Such modifications can be made provided that there are suitable administrative controls over these changes. These details, that are presently included in TS and which are removed by this amendment, do not constitute performance requirements necessary to ensure safe operation of the facility and, therefore, do not warrant being included in TS. At the same time, suitable administrative controls ensure that there will be careful review and analysis by competent individuals of any changes in the Fire Protection Program including those technical and administrative requirements removed from the TS to ensure that nuclear safety is not adversely affected. These controls include: (1) the TS administrative controls that are applicable to the Fire Protection Program, (2) the license condition on implementation of, and subsequent changes to, the Fire Protection Program, and (3) the 10 CFR 50.59 criteria for evaluating changes to the Fire Protection Program as described in the FSAR.

The specific details relating to fire protection requirements removed from TS by this amendment include those specifications for fire detection systems, fire suppression systems, fire barriers, and fire brigade staffing requirements. The administrative control requirements have been modified to include Fire Protection Program implementation as an element for which written procedures must be established, implemented, and maintained. In addition, the audit responsibilities of the On-Site Review and Investigative Function were expanded to include the review of the Fire Protection Program and implementing procedures and submittal of recommended changes to the Off-Site Review and Investigative Function.

The TS changes proposed by the licensee are in accordance with the guidance provided by Generic Letter 88-12, as addressed in the items below.

(1) Specification 6.1.G.2.a.13, On-Site Review and Investigative Function, was revised to add the review of the fire protection program implementation and Specification 6.1.G.1.a was revised to include the review of recommended changes by the Off-Site Review and Investigative Function.

(2) With the inclusion of Specification 6.1.G.1.a(11), Fire Protection Program implementation has been added to those programs for which written procedures are required. Specification 6.1.H, which was approved in a previous amendment, contains an inspection and audit requirement for the Fire Protection Program.

(3) Specifications 3.12 and 4.12, Fire Suppression Systems, their associated Surveillance Requirements, and Bases (including Fire Barriers and Fire Detection Instrumentation, and their associated Surveillance Requirements and Bases) were removed.

(4) Specification 6.2.C on fire brigade staffing requirements was removed.

As required by Generic Letter 86-10, the licensee by letter dated June 20, 1989 confirmed that the NRC approved Fire Protection Program has been incorporated into the FSAR and stated that any other references determined to be appropriate would be included in the 1989 calendar year FSAR update. Also, the licensee has proposed that the existing licensing conditions on the Fire Protection Program be replaced with the standard condition noted in Generic Letter 86-10.

The licensee confirmed that the operational conditions, remedial actions, and test requirements associated with the fire protection TS will be approved prior to removal from the TS and within 60 days from the approval date of this safety evaluation and included in the Fire Protection Program incorporated into the UFSAR. This satisfies the guidance of Generic Letter 88-12.

On the basis of its review of the above items, the staff concludes that the licensee has met the guidance of Generic Letter 88-12. Therefore, the staff finds the proposed changes acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes to the use of the facility components located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released off site, and that there is no significant increase in individual or cumulative occupational exposure. The staff has determined that the amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

5.0 CONCLUSION

The Commission made proposed determinations that the amendments involve no significant hazards consideration, which were published in the Federal Register (54 FR 13762) on April 5, 1989. The Commission consulted with the

State of Illinois. No public comments were received, and the state of Illinois did not have any comments.

On the basis of the considerations discussed above, the staff concludes that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Dennis J. Kubicki, SPLB/DEST
Thomas G. Dunning, OTSB/DOEA
Byron L. Siegel, NRR/DRSP

Dated: June 30, 1989

Tab 3

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

AUG 14 1988

August 9, 1989

Docket Nos. 50-237 and 50-249

Mr. Thomas J. Kovach
Nuclear Licensing Manager
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Kovach:

SUBJECT: CORRECTION TO THE INDEX PAGES FOR DRESDEN UNITS NO'S 2 AND 3 FOR AMENDMENT NOS. 106 and 101 (TAC NOS. 71256 AND 71257)

The purpose of this letter is to notify you that the index pages for these amendments contain two errors. The pages identified as 6-7 and 6-13 to be removed should be changed to pages 6-5 and 6-11 as shown in the enclosed index pages. These changes are necessary to correct the original page numbers which were superseded by a subsequent amendment.

Sincerely,

Byron L. Siegel

Byron L. Siegel, Project Manager
Project Directorate III-2
Division of Reactor Projects III,
IV, V, and Special Projects

Enclosures:
As stated

cc w/enclosures: See next page

Mr. Thomas J. Kovach
Commonwealth Edison Company

Dresden Nuclear Power Station
Units 2 and 3

cc:

Michael I. Miller, Esq.
Sidley and Austin
One First National Plaza
Chicago, Illinois 60603

Mr. J. Eenigenburg
Plant Superintendent
Dresden Nuclear Power Station
Rural Route #1
Morris, Illinois 60450

U. S. Nuclear Regulatory Commission
Resident Inspectors Office
Dresden Station
Rural Route #1
Morris, Illinois 60450

Chairman
Board of Supervisors of
Grundy County
Grundy County Courthouse
Morris, Illinois 60450

Regional Administrator
Nuclear Regulatory Commission, Region III
799 Roosevelt Road, Bldg. #4
Glen Ellyn, Illinois 60137

Mr. Michael E. Parker, Chief
Division of Engineering
Illinois Department of Nuclear Safety
1035 Outer Park Drive, 5th Floor
Springfield, Illinois 62704

ATTACHMENT TO LICENSE AMENDMENT NO. 106

PROVISIONAL OPERATING LICENSE DPR-19

DOCKET NO. 50-237

Revision 8
April 1992

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
iv	iv
3/4.12-1	
3/4.12-2	
3/4.12-3	
3/4.12-4	
3/4.12-5	
3/4.12-6	
3/4.12-7	
3/4.12-8	
3/4.12-9	
3/4.12-10	
3/4.12-11	
3/4.12-12	
3/4.12-13	
3/4.12-14	
B 3/4.12-15	
B 3/4.12-16	
B 3/4.12-17	
B 3/4.12-18	
B 3/4.12-19	
B 3/4.12-20	
B 3/4.12-21	
6-1	6-1
6-5	6-5
6-11	6-11

ATTACHMENT TO LICENSE AMENDMENT NO. 101

FACILITY OPERATING LICENSE DPR-25

DOCKET NO. 50-249

Revision 8
April 1992

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
iv	iv
3/4.12-1	
3/4.12-2	
3/4.12-3	
3/4.12-4	
3/4.12-5	
3/4.12-6	
3/4.12-7	
3/4.12-8	
3/4.12-9	
3/4.12-10	
3/4.12-11	
3/4.12-12	
3/4.12-13	
3/4.12-14	
B 3/4.12-15	
B 3/4.12-16	
B 3/4.12-17	
B 3/4.12-18	
B 3/4.12-19	
B 3/4.12-20	
B 3/4.12-21	
6-1	6-1
6-5	6-5
6-11	6-11

Tab II

Fire Protection Report

Dresden Administrative Technical Requirements (DATR's) for Fire Protection

The Fire Protection and Safe Shutdown DATR's are available through Central Files.

Tab III

DRESDEN 2&3

FIRE PROTECTION PROGRAM DOCUMENTATION PACKAGE

NRC Inspection Reports

Tab	Title
1	May 25, 1984 Inspection Reports No. 50-010/84-01, 50-237/84-06; 50-249/84-05
2	July 3, 1984 Inspection Report No. 50-010/84-09, 50-237/84-11; 50-249/84-10
	July 30, 1984 CECo letter from D. L. Farrar to J. G. Keppler (NRC); Response to Inspection Report No. 50-010/84-09, 50-237/84-11, 50-249/84-10
3	November 14, 1985 Inspection Report No. 50-237/85033, 50-249/85029
	December 26, 1985 Notice of Violation Concerning Inspection Reports No. 50-237/85033, 50-249/85029
	January 24, 1986 CECo letter from D. L. Farrar to J. G. Keppler (NRC) transmitting response to Inspection Report No. 50-237/85033 and 50-249/85029
4	February 26, 1986 Inspection Report No. 50-249/86006
	May 6, 1986 (CECo) letter from D. L. Farrar to J. G. Keppler (NRC) transmitting the response to Inspection Report 50-249/86006
	July 17, 1986 (CECo) letter from D. L. Farrar to J. G. Keppler (NRC) discussing Inspection Report No. 50-249/86006
5	December 21, 1987 Inspection Report No. 50-237/87035 and 50-249/87034
6	December 14, 1987 Inspection Report No. 50-237/87037 and 50-249/87036
7	January 3, 1989 Inspection Report No. 50-237/88010 and 50-249/88012 to assess compliance with 10 CFR 50 Appendix R
	Appendix R Audit Questions April 18-22, 1988
	February 1, 1989 CECo letter from H. E. Bliss to A. Bert Davis transmitting the response to Inspection Report No. 50-237/88010 and 50-249/88012

DRESDEN 2&3

FIRE PROTECTION PROGRAM DOCUMENTATION PACKAGE

NRC Inspection Report (Cont'd)

Tab	Title
8	January 23, 1989 Inspection Report No. 50-237/88030 and 50-249/88031 to review deficiencies in fire wrap installations and training to installers
	April 14, 1989 CECo letter from E. D. Eenigenberg to R. J. Israelson (3M) on review of installed E-50 Fire Wrap Removable Covers
	May 3, 1989 letter from R. J. Israelson (3M) to E. D. Eenigenberg (CECo), response to April 14, 1989 letter
9	February 28, 1989 Inspection Report 50-249/89004 concerning the June 4, 1988 fire in the Drywell Expansion Gap
10	April 14, 1989 Inspection Report 50-237/89008 and 50-249/89009 reviewing allegations regarding unsealed openings inside conduits in fire walls
11	June 9, 1989 Inspection Report 50-237/89013 and 50-249/89012 to review implementation of the licensee's fire protection program
12	July 31, 1989 Inspection Report 50-010/89002, 50-237/89017 and 50-249/89016
13	December 26, 1989 Inspection Report No. 50-237/89022 and 50-249/89021
	January 25, 1990 CECo letter from T. J. Kovach to A. Bert Davis (NRC), Response to Notice of Violation and Inspection Report No. 50-237/89022 and 50-249/89021
14	August 24, 1990 Inspection Report No. 50-237/90017 and 50-249/90017
	September 24, 1990 CECo letter from T. J. Kovach to A. Bert Davis (NRC), Response to Notice of Violation and Inspection Report No. 50-237/90017 and 50-249/90017
	November 28, 1990 NRC letter from A. Bert Davis to C. Reed (CECo) proposed Imposition of Civil Penalty

DRESDEN 2&3

FIRE PROTECTION REPORT

NRC Inspection Reports (Cont'd)

Tab	Title
15	December 7, 1990 Inspection Report No. 50-237/90023 and 50-249/90003.
	December 14, 1990 CECo letter from T.J. Kovach to A. Bert Davis (NRC) discussing unresolved Item 50-237/9002-06 in Inspection Report No. 50-237/90023 and 50-249/90023.
	January 7, 1991 CECo letter form T.J. Kovach to A. Bert Davis (NRC), response to Notice of Violation contained in Inspection Report No. 50-237/90023 and 50-249/90023.
	February 6, 1991 NRC letter from H.J. Miller to C. Reed (CECo) responding to CECo's letter of January 7, 1991.
16	January 7, 1991 Inspection Report No. 50-237/90027 and 50-249/90026.
	February 15, 1991 CECo letter from T.J. Kovach and A. Bert Davis (NRC). Response to Notice of Violation Associated with Inspection Report No. 50-237/90027 and 50-237/90026.
17	March 15, 1991 Inspection Report No. 50-237/91004 and 50-249/91004 to review implementation of the routine fire protection program.
	March 27, 1991 CECo letter from T.J. Kovach to A. Bert Davis (NRC). Response to Notice of Violation Associated with Inspection Report No. 50-237/91004 and 50-249/91004.
18	March 2, 1993 Inspection Report No. 50-237/93002 and 50-249/93002.
19	May 20, 1996 Inspection Report Nos. 50-010/96002, 50-237/96002 and 50/249/96002.
20	November 14, 1996 Inspection Report Nos. 50-237/96012, 50-249/96012, 50-254/96016, and 50-265/96016
	December 12, 1996 ComEd letter from E.S. Kraft to NRC, response to apparent violation contained in Inspection Report Nos. 50-237/96012, 50-249/96012, 50-254/96016, and 50-265/96016.

DRESDEN 2&3

FIRE PROTECTION REPORT

NRC Inspection Reports (Cont'd)

Tab	Title
	December 20, 1996 ComEd letter from J. B. Hosmer to NRC, supplemental response to apparent violation contained in Inspection Report Nos. 50-237/96012, 50-249/96012, 50-254/96016 and 50-265/96016.
	March 6, 1997 ComEd letter from J. B. Hosmer to NRC, regarding protection of motor operated valves during postulated hot shorts.
21	March 6, 1998 Inspection Report Nos. 50-237/97021 and 50-249/97021.
	April 6, 1998 ComEd letter from J.M. Heffley to NRC, response to Notice of Violation contained in Inspection Report Nos. 50-237/97021 and 50-249/97021.
22	December 18, 1998 Inspection Report Nos. 50-237/98029 and 50-249/98029.
23	June 19, 2002 Inspection Report Nos. 50-237/02-06(DRS) and 50-249/02-06(DRS).
24	May 5, 2005 Inspection Report Nos. 05000237/2005002(DRS) and 05000249/2005002(DRS)

Tab 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

MAY 25 1984

MAY 30 REC'D

Docket No. 50-10
Docket No. 50-237
Docket No. 50-249

Commonwealth Edison Company
ATTN: Mr. Cordell Reed
Vice President
Post Office Box 767
Chicago, IL 60690

Gentlemen:

This refers to the routine safety inspection conducted by T. M. Tongue, S. Stasek, C. D. Anderson, and R. Lanksbury of this office on March 27, 1984 through May 21, 1984, of activities at Dresden Nuclear Power Station, Units 1, 2 and 3 authorized by NRC Operating Licenses No. DPR-02, DPR-19, and DPR-25, and to the discussion of our findings with Mr. D. Scott and others of your staff at the conclusion of the inspection.

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.

No items of noncompliance with NRC requirements were identified during the course of this inspection.

In accordance with 10 CFR 2.790(a), a copy of this letter and the enclosure(s) will be placed in the NRC Public Document Room unless you notify this office, by telephone, within ten days of the date of this letter and submit written application to withhold information contained therein within thirty days of the date of this letter. Such application must be consistent with the requirements of 2.790(b)(1). If we do not hear from you in this regard within the specified periods noted above, a copy of this letter and the enclosed inspection report will be placed in the Public Document Room.

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

Sincerely,

N. D. Shafer, Chief
Projects Branch 2

Enclosure: Inspection Reports
No. 50-010/84-01(DPRP);
No. 50-237/84-06(DPRP);
No. 50-249/84-05(DPRP)

cc w/encl:
D. L. Farrar, Director
of Nuclear Licensing
D. J. Scott, Station
Superintendent
DMB/Document Control Desk (RIDS)
Resident Inspector, RIII
Phyllis Dunton, Attorney
General's Office, Environmental
Control Division

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-010/84-01(DPRP); 50-237/84-06(DPRP); 50-249/84-05(DPRP)

Docket Nos. 50-010; 50-237; 50-249 Licenses No. DPR-02; DPR-19; DPR-25

Licensee: Commonwealth Edison Company
P. O. Box 767
Chicago, IL 60690

Facility Name: Dresden Nuclear Power Station, Units 1, 2, and 3

Inspection At: Dresden Site, Morris, IL

Inspection Conducted: March 27, 1984 through May 21, 1984

Inspectors: T. M. Tongue

S. Stasek

C. D. Anderson

R. Lanksbury

Approved By: N. J. Chrissotimos, Chief
Projects Branch 2C

5 25 84
Date

Inspection Summary

Inspection during the period of March 27 through May 21, 1984 (Reports No. 50-10/84-01(DPRP); 50-237/84-06(DPRP); 50-249/84-05(DPRP))

Areas Inspected: Routine unannounced resident inspection of action on previous inspection findings, regional requests, 10 CFR 21 notifications, operational safety, events, fire protection program, surveillance, maintenance, IE Bulletins, licensee event reports, spent fuel shipments, Three Mile Island modifications, regulatory performance improvement plan, Unit 1 chemical cleaning, independent inspection, report review, and meeting with local municipal officials. The inspection involved a total of 398 inspector-hours onsite by 4 NRC inspectors including 78 inspector-hours onsite during off-shifts.

Results: Of the 16 areas inspected, no items of noncompliance or deviations were identified.

DETAILS

SECTION I

1. Persons Contacted

Commonwealth Edison - Station Personnel

*D. Scott, Station Superintendent
R. Ragan, Operations Assistant Superintendent
J. Eenigenburg, Maintenance Assistant Superintendent
*J. Wujciga, Administrative and Support Services Assistant Superintendent
J. Brunner, Technical Staff Supervisor
R. Christensen, Unit 1 Operating Engineer
*J. Almer, Unit 2 Operating Engineer
T. Ciesla, Unit 3 Operating Engineer
D. Sharper, Waste Systems Engineer
*G. Myrick, Radiation Chemistry Supervisor
B. Saunders, Station Security Administrator
L. Williams, Quality Assurance Coordinator
*R. Stobert, Quality Assurance Inspector
R. Stols, Quality Assurance Engineer
*T. Gilman, Chemistry Supervisor
*S. McDonald, Radiation Protection Supervisor
M. Dillon, Fire Marshal
T. Ziakis, Emergency Preparedness Coordinator
D. Ambler, Health Physicist

Commonwealth Edison - Corporate Personnel

D. A. Adam, Lead Health Physicist Field Services Engineer

Contractors

Home Transportation Company

K. Jones, Driver

Coyne Industrial Laundry - Joliet

E. Kasmark, General Manager

NRC Personnel
Region III

R. Paul, Health Physicist
A. Januska, Health Physicist
G. France, Health Physicist
R. Lickus, Director, State and Government Affairs

The inspectors also talked with and interviewed several other licensee employees, including members of the technical and engineering staffs, reactor and auxiliary operators, shift engineers and foremen, electrical, mechanical and instrument personnel, and contract security personnel.

*Denotes those attending one or more exit interviews conducted on April 19, April 23, May 4, May 8, and May 21, 1984, and informally at various times throughout the inspection period.

2. Action on Previous Inspection Findings

(Closed) Open Item (237/75-01(DPRP)): Torus (suppression pool) baffle removed. The baffles were removed in accordance with the Mark I containment modifications program.

(Closed) Open Item (237/76-06(DPRP)): Traversing Incore Probe (TIP) isolation ball valve. This item is being reviewed generically. If any action is required, it will be forthcoming in appropriate NRC correspondence.

(Closed) Open Item (249/77-01(DPRP)): High Pressure Coolant Injection (HPCI) system motor operated valve wiring change to prevent cycling after closing. The inspector reviewed modification packages and, through interviews with station personnel, verified that Unit 3 HPCI valves MO 2301-3, 5, 8 and 9 had wiring and motor operator modifications to prevent hammering and damage.

(Closed) Open Item (237/77-145(DPRP)): Control of Licensee Offsite Work. Licensee offsite work has been reviewed on numerous occasions directly and indirectly. This was done through Division of Engineering routine inspections, management appraisal team audit and a performance appraisal team inspection. In each case all inspection findings have been appropriately addressed by the NRC and the licensee.

(Closed) Open Item (237/78-07-02(DPRP) and 249/78-07-02(DPRP)): Long term corrective action on improper 125 volt D.C. cable separation. The licensee has replaced the affected cables and has installed an additional battery charger.

(Closed) Open Item (249/78-25-04(DPRP)): Small leakage identified in the 3A, B, and C reactor feedpump minimum flow lines. The licensee completed the long term corrective actions by replacing the flow control valves, installing smaller orifices, and replacing the eroded pipe elbows with harder chrome alloy steel.

(Closed) Open Item (249/79-01-01(DPRP)): Leak on main steam line control valve above seat drain line. The licensee replaced the affected 3000 psi steel line components with 6000 psi steel components.

(Closed) Open Item (237/79-13-02(DPRP) and 249/79-11-02(DPRP)): Torus drain isolation to inhibit inadvertent torus draining to main condenser. The licensee has completed installation of extra valving in the line used to pump excess water inventory in the torus to the condenser hotwell. Valves 2(3)-1599-61 and 2(3)-1599-62 are located downstream of the Low Pressure Coolant Injection (LPCI) System Cross-Tie line on both Units 2 and 3. The inspector reviewed the associated modification package and verified completion of installation of the aforementioned valves and found all to be acceptable.

(Closed) Open Item (249/79-18-01(DPRP)): Replacement of G.E. CR-120A relays. The relays were replaced as stated and in accordance with the licensee's modification program.

(Closed) Unresolved Item (10/79-19-03(DPRP), 237/79-23-03(DPRP), and 249/79-21-01(DPRP)): Licensee organizational changes were not shown in Technical Specifications. This matter was also identified during reviews of Three Mile Island modifications (NUREG-0737 Item I.A.1.3.2.A) and by letter dated June 4, 1982, the licensee stated that modifications were implemented as required and a change request to the Technical Specifications had been submitted.

(Closed) Noncompliance (10/80-18-01 (Region V Inspection)): Radwaste drum lid improperly secured; identified at the Richland, Washington disposal site. The licensee has improved procedures for closing radwaste containers, required a management person to verify the condition of all shipments, and is showing more quality assurance and quality control attention to these shipments. There have been no additional findings since this citation.

(Open) Unresolved Items (237/81-09-06(DE) and 249/81-06-06(DE)): Not all fire brigade members participate in at least two drills per year; (237/81-09-07(DE) and 249/81-06-07(DE)). Fire brigade training does not include the use of preplans or strategies for specific instruction and reference during an emergency. By memo dated December 28, 1983, a request was resubmitted to NRC headquarters for guidance.

(Closed) Open Item (237/81-20-01(DPRP) and 249/81-14-01(DPRP)): Safety and relief valve acoustical monitors - environmental and seismic certification. The licensee has submitted information to NRR concerning the environmental and seismic qualifications of the installed system in accordance with NUREG-0737 Task Action Item II.D.3. NRR currently has the licensee's submittals under review. This item will therefore be followed under Task Action Item II.D.3.

(Closed) Noncompliance (237/81-24-02(DPRP)): High Pressure Coolant Injection (HPCI) inoperable due to maintenance on the steam stop valve without a proper procedure. The licensee has developed a more detailed maintenance manual for the steam stop valve. In addition, the licensee is developing a permanent maintenance procedure for this activity.

(Closed) Unresolved Item (237/81-35-01(DE) and 249/81-27-01(DE)): Conflicting statement by equipment operator related to event of November 25, 1981, regarding starts, stops and associated alarms of the emergency diesel-generator circulating water pumps. This was resolved through investigation by the licensee and reported to the NRC via letter dated December 4, 1981.

(Closed) Unresolved Item (237/81-35-02(DPRP) and 249/81-27-02(DPRP)): Possible violation of General Design Criteria 44 of 10CFR50 Appendix A in regard to common suction modes for the diesel generator cooling water pumps (DGCWPs). Via a telephone conversation with NRR (as documented in a memo dated June 18, 1982), the common suction mode of operation for the DGCWPs was deemed to be acceptable as long as it was used only on a temporary basis (such as for required maintenance), and procedures were in place to ensure that a water source was available to the pumps. The licensee currently has Dresden Operating Procedure (DOP) 4400-5 in place that addresses the implementation of these requirements. Moreover, following discussions with the inspector, the licensee has initiated further changes to the procedure to better clarify the aforementioned requirements.

(Closed) Open Item (237/81-37-07(DPRP) and 249/81-29-07(DPRP)): Inspect pilot valve junction box interiors on electromatic relief valves to assure there is no interference with electrical contacts. This was completed on Units 2 and 3 and no further problems were identified.

(Closed) Open Item (237/81-38-02(DPRP) and 249/81-31-02(DPRP)): Access covers left open on components following maintenance. The licensee has issued a memo to all departments at Dresden stating the importance of restoring equipment following maintenance and surveillance.

(Closed) Noncompliance (237/82-10-02(DPRP) and 249/82-11-02(DPRP)): Improper usage of general purpose hoses. The licensee has modified Dresden Administrative Procedure (DAP) 3-7 to reflect requirements for proper hose usage onsite. Also, modifications have been made so that only breathing air hoses use special snap-tite connectors. All other general purpose hoses utilize Chicago type fittings.

(Closed) Noncompliance (237/82-18-01(DE) and 249/82-19-01(DE)): Inadequate fire protection surveillances. The licensee has added the Cardox manual valve to the surveillance list and corrected the control room smoke detector surveillance lists.

(Closed) Noncompliance (237/82-20-01(DPRP) and 249/82-21-01(DPRP)): Measures were not established to control nonconforming parts in order to prevent their inadvertent installation or use. The licensee identified where all of the nonconforming valve guides were used and replaced the only nonconforming component used on Unit 3 during an October 1982 outage. Dresden Administrative Procedures, DAP 11-4 and 11-5, were implemented for classification and evaluation of spare parts used in safety related applications.

(Closed) Noncompliance (237/82-20-02(DPRP) and 249/82-21-02(DPRP)): Measures were not established to distribute safety classifications to appropriate corporate or onsite personnel to assure the procurement, installation, and use of quality parts. Station Nuclear Engineering Department (SNED) revised procedure Q.12 to assure correspondence on classification and listings of safety-related items is distributed to the site, corporate quality assurance and respective project groups.

(Closed) Noncompliance (237/82-22-01(DPRP)): Failure to maintain primary containment integrity per Technical Specification requirements. The licensee has completed all modifications, procedure changes, and operator retraining concerning both Units 2 and 3 torus sightglass operation per commitments.

(Closed) Noncompliance (237/82-22-02(DPRP)): Failure to report containment integrity violation in a timely manner. All clerks responsible for operating the telefax machines have been retrained concerning the importance of NRC notification requirements and the licensee's procedures concerning the telefaxing of these notifications.

(Closed) Open Item (237/82-23-01(DPRP) and 249/82-23-01(DPRP)): Process computer alarm disabled, bypasses operator acknowledgement. The licensee has installed new computer panels within reasonable reach of the operator, reviewed and removed some computer alarms, and implemented a shift surveillance to identify bypassed alarms.

(Closed) Open Item (10/82-17-02(EPS); 237/82-24-02(EPS); and 249/82-24-02(EPS)): Inadequate hydrological forecasting exists at the site and load dispatcher's office. The licensee has an Army Corps of Engineers letter dated June 21, 1983, confirming a procedure to inform Dresden of best estimates on crest forecasts at the Dresden Lock and Dam. In addition, by letter dated June 9, 1983, the licensee has expanded their contract with a weather forecasting agency verifying the willingness to respond to the station needs during an emergency.

(Closed) Noncompliance (249/82-28-01(DE)): Failure of the Dresden Onsite Review Committee to provide an adequate review of a procedure. The inspector reviewed the licensee's response to this item, discussed it with the licensee and concluded that their contentions of no additional corrective measures being required appears to be valid. Based on this and the currently inplace measures to ensure that the station procedures for calculating core thermal power are properly reviewed prior to their use, the inspector has no further concerns in this area.

(Closed) Open Item (249/82-28-02(DE)): Lack of signoff/dating blocks on checklists in some startup physics test procedures. The inspector reviewed the startup physics test procedures and noted that the licensee had added additional signoff/dating blocks in the procedure where they deemed it appropriate to do so. The inspector did not note any additional areas where it would appear appropriate to have signoff/dating blocks and therefore has no further concerns in this area.

(Closed) Open Item (237/83-11-09(DPRP) and 249/83-09-09(DPRP)): Develop a training module and train maintenance mechanics on the proper use of sealants and lubricants. Maintenance mechanics were trained within the prescribed commitment and a permanent training module was developed by the Production Training Center for training new employees.

(Closed) Open Item (237/83-11-12(DPRP) and 249/83-09-12(DPRP)): Torus (suppression pool) internals where modifications had no paint (preservation) on newly welded areas. The licensee has a plan for periodic repainting torus internals during subsequent outages.

(Closed) Noncompliance (237/83-14-01(DPRP) and 249/83-13-01(DPRP)): Failure to restore systems to normal following maintenance and/or surveillance. The licensee has issued a memo to all departments at Dresden to assure the adequacy of housekeeping practices for restoration of systems and components to prevent the intrusion of substances that could result in equipment failures.

(Closed) Noncompliance (237/83-14-02(DPRP) and 249/83-13-02(DPRP)): Inadequate corrective action from a previous event resulted in one train of an emergency core cooling system being inoperable. The licensee modified maintenance procedure DMP 040-6, "Safety Related Motor Operated Valves - Data and Settings", given specific instructions to appropriate personnel and distributed copies of deviation reports to Maintenance Departments for greater awareness.

(Closed) Open Item (10/83-12-01(DPRP), 237/83-20-01(DPRP), and 249/83-18-01(DPRP)): Licensee on-the-job training for maintenance personnel needs to be upgraded and documented. The licensee is implementing a more formal, four to five year program for mechanics. This was implemented in mid-1983 for mechanical and electrical mechanics, and is scheduled to be implemented for instrument mechanics by September 1984.

(Closed) Open Item (10/83-12-02(DPRP), 237/83-20-02(DPRP), and 249/83-18-02(DPRP)): New procedure and procedure modification backlog needs to be reduced, maintenance procedures need greater detail, and development of maintenance manuals should be stepped up. The licensee has developed a formal review process to help reduce the backlog, maintenance procedures are being developed with more attention to details, and more maintenance manuals are being developed or modified to help assure maintenance with better control over the work. In addition, the licensee is converting a number of maintenance manuals to approved maintenance procedures.

(Closed) Open Item (10/83-12-03(DPRP), 237/83-20-03(DPRP), and 249/83-18-03(DPRP)): Work packages on safety related valves needs a generic set of Quality Assurance/Quality Control (QA/QC) hold points. In addition, work could start without appropriate QA or QC approval. The licensee has made the following changes:

3. Modified Dresden Administrative Procedure DAP 15-1 "Work Requests", by adding a set of guidelines on QC hold and witness points and generated DAP 15-3, "Preparation of Safety Related or Reliability Related Work Packages on Off-shifts", which requires QC and/or QA approval to start safety related work.

(Closed) Open Item (10/83-12-04(DPRP), 237/83-20-04(DPRP), and 249/83-18-04(DPRP)): Equipment and parts obtained as nonsafety related must be upgraded prior to use in a safety related application. The licensee has modified Dresden Administrative Procedures DAP 11-4, "Control of the Classification List of Safety Related (SR), Non-safety Related (NSR) and American Society of Mechanical Engineering (ASME) Code-Related Systems, Structures and Components"; DAP 11-5 Classification of Non-Safety Related (NSR) Subcomponents/Parts Used on/in Safety Related (SR) Systems, Structures and Components"; DAP 11-6, "Request for Purchase and Receiving Inspection Guidelines"; and DAP 11-7, "Technical Evaluation of Parts Used in Safety Related Components."

These were reviewed by the inspector and appear to be acceptable to correct the problem identified. More recently problems with code related drywell-to-torus vacuum breaker shaft seals is being addressed under a separate special inspection.

(Closed) Open Item (10/83-12-05(DPRP), 237/83-20-05(DPRP), and 249/83-18-05(DPRP)): Improved communications needed between maintenance operations and radiation protection. In addition to previous corrective actions identified, an ALARA review is required on work request forms. Review by the inspector shows that the corrective actions have been successful.

(Closed) Open Items (249/83-19-01(DPRP), 249/83-19-02(DPRP), 249/83-19-03(DPRP), and 249/83-19-04(DPRP)): NRC Order of August 26, 1983, related to cracks identified in BWR large diameter piping. The licensee implemented leakage control requirements, shutdown and examined all piping as required, and completed the remainder of the items as required. By NRR letter dated March 15, 1984, the order has been rescinded to allow continued operation within some constraints.

(Closed) Noncompliance (237/83-21-01(DPRP)): Insufficient corrective action relative to holes in torus-to-drywell vacuum breaker lines. Following the second incident on August 11, 1983, the licensee instituted more intensive corrective actions (as outlined in a letter dated November 10, 1983) to ensure further incidents of this type would not occur at Dresden. Modifications to the piping supports for the torus-to-drywell vacuum breaker lines have since been completed for both Units 2 and 3 without further incident.

(Closed) Noncompliance (237/83-21-03(DPRP)): Valve misalignment due to inadequate procedures. The licensee has modified the identified procedures (DOS 1500-1 and DOS 1600-1) to include more specific instructions for correctly draining between containment spray valves 1501-27A(B) and 1501-28A(B) during valve operability surveillances.

No further items of noncompliance or deviations were identified.

for those components. In addition, the inspector verified that the licensee and the supplier were aware of this issue for identification of other components that may be supplied in the future.

No items of noncompliance or deviations were identified.

5. Operational Safety Verification

The inspectors observed control room operations, reviewed applicable logs and conducted discussions with control room operators during the period from March 27 to May 21, 1984. The inspectors verified the operability of selected emergency systems, reviewed tagout records and verified proper return to service of affected components. Tours of Unit 2 reactor building and turbine buildings were conducted to observe plant equipment conditions, including potential fire hazards, fluid leaks, and excessive vibrations and to verify that maintenance requests had been initiated for equipment in need of maintenance.

During the inspection period while Unit 3 was in an outage to repair the main turbine, the inspectors verified that surveillance tests were conducted, containment integrity requirements were met, and emergency systems were available as necessary.

Throughout the entire inspection period, Unit 1 remained in a longterm shutdown condition with all fuel removed from the vessel. The inspectors verified that all applicable requirements for Unit 1 were met during this period.

The inspectors, by observation and direct interview, verified that the physical security plan was being implemented in accordance with the station security plan.

The inspectors observed plant housekeeping/cleanliness conditions and verified implementation of radiation protection controls. During the inspection, the inspector walked down the accessible portions of the following systems to verify operability by comparing system lineup with plant drawings, as-built configuration or present valve lineup lists; observing equipment conditions that could degrade performance; and verified that instrumentation was properly valved, functioning, and calibrated.

a. Unit 2

Low Pressure Coolant Injection System (both loops), Core Spray System (both loops), Isolation Condenser, Unit 2 Emergency Diesel Generator, and portions of the Control Rod Drive System.

b. Unit 3

Unit 3 Emergency Diesel Generator

an upper bound. (NRC 10 CFR 50.72 reporting requirements sets 2 MPC as the lower limit that need be reported.) Followup review of this incident has been assigned to the Region III Facilities Radiation Protection Section (FRPS) to be looked at during their next inspection at Dresden.

No items of noncompliance or deviations were identified.

7. Fire Protection Program

During the inspection period, the inspector reviewed the licensee's fire protection program against Technical Specification requirements and licensee commitments. Walkdowns were conducted of the accessible portions of the cardox system, halon system, water suppression system, and portable fire protection equipment. The following surveillances were reviewed for adequacy and completeness:

DFPP 4114-2 Reactor Building Monthly Fire Equipment Inspection
DFPP 4114-3 Turbine Buildino Monthlv Fire Eouioment Insoection
DFPP 4145-1 Cardox System Semi-Annual Maintenance Test
DFPP 4153-2 Emergency Lighting Monthly Inspection
DFPP 4185-2 Smoke Detector Semi-Annual Maintenance Test

No items of noncompliance or deviations were identified in this area.

8. Surveillance Observation

The inspectors observed Technical Specifications required surveillance testing on the Unit 2 Emergency Diesel Generator and verified that testing was performed in accordance with adequate procedures, that test instrumentation was calibrated, that limiting conditions for operation were met, that removal and restoration of the affected components were accomplished, that test results conformed with Technical Specifications and procedure requirements and were reviewed by personnel other than the individual directing the test, and that any deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel.

The inspectors also witnessed/reviewed portions of the following test activities:

a. Unit 2

Local Power Range Monitors Calibration, Core Spray System Pump Test with Torus Available, Low Pressure Coolant Injection System Valve Operability Test, and High Pressure Coolant Injection System Motor Operated Valves and Pump Operability Test.

b. Unit 3

Low Pressure Coolant Injection System Valve Operability Test

No items of noncompliance or deviations were identified in this area.

9. Maintenance Observation

Station maintenance activities of safety related systems and components listed below were observed/reviewed to ascertain that they were conducted in accordance with approved procedures, regulatory guides and industry codes or standards and in conformance with Technical Specifications.

The following items were considered during this review: the limiting conditions for operation were met while components or systems were removed from service; approvals were obtained prior to initiating the work; activities were accomplished using approved procedures and were inspected as applicable; functional testing and/or calibrations were performed prior to returning components or systems to service; quality control records were maintained; activities were accomplished by qualified personnel; parts and materials used were properly certified; radiological controls were implemented; and, fire prevention controls were implemented.

Work requests were reviewed to determine status of outstanding jobs and to assure that priority is assigned to safety related equipment maintenance which may affect system performance.

The following maintenance activities were observed/reviewed:

a. Unit 2

'C' Traversing Incore Probe Machine Maintenance and Emergency Diesel Generator Quarterly Maintenance

b. Unit 3

'A' Low Pressure Coolant Injection Heat Exchanger Cleaning

No items of noncompliance or deviations were identified in this area.

10. IE Bulletin Followup

Each of the following IE Bulletins were reviewed by the resident inspector to determine if: (1) the licensee's written response was submitted within the time limitations stated in the bulletin, (2) the written response included all information required to be reported, (3) the written response included adequate corrective action commitments based on information presented in the bulletin and the licensee's response, (4) licensee management forwarded copies of the written response to the required onsite management representatives, (5) information discussed in the licensee's response was accurate, and (6) the corrective action taken was as described in the response.

(Closed) IEB 83-07 Apparent Fraudulent Products Sold by Ray Miller, Inc.

(Closed) IEB 83-08 Electrical Circuit Breakers With an Under-Voltage Trip Feature in Use in Safety-Related Applications Other Than the Reactor Trip System.

No items of noncompliance or deviations were identified in this area.

of Transportation (the responsible agency) was informed of the event. The licensee examined and repaired the trailer in accordance with applicable codes per DOT and placed it back in service.

No items of noncompliance or deviations were identified in this area.

13. Three Mile Island Modifications

The inspector reviewed the following TMI items for development and implementation per NUREG-0737 requirements and licensee commitments.

II.F.1.3: Accident Monitoring - Containment High Range

In response to this task action item, the licensee, as documented via a letter dated April 15, 1982, installed containment radiation monitors on both Units 2 and 3. Further licensee commitments specified that associated procedures, when finalized, would include appropriate correction factors to modify instrument readings to correspond with actual containment radiation levels. On June 1, 1982, the information concerning these correction factors and how they were arrived at were submitted for review. Verification of licensee actions concerning this task item was assigned to the Region III Facilities Radiation Protection Section. As documented in Inspection Report (237/82-30; 249/82-31(DRMSP)), the licensee appears to meet the intent of the NUREG-0737 requirements for this item. Therefore II.F.1.3 is considered closed at this time.

III.A.2.4: Installation of Emergency Response Facility (ERF) Meteorological Hardware and Software.
III.A.2.5: Full Operability of III A.2.4.
III.A.2.6: Review of Dose Calculation Methodology (DCM) By The Licensee.
III.A.2.8: Full Operation of Class B Model.

These task action items, as currently outlined, reflect requirements as issued in NUREG-0660 and NUREG-0737. However, since their issuance, Secy 82-111 "Requirements for Emergency Response Capability" has been issued which significantly modified the original requirements. As documented in a memo dated March 1, 1984 from C. Paperiello to C. Norelius, the Emergency Response Facility (ERF) Appraisal Program is the current program proposed for the review of these items. Since these task action items will be reviewed using the Secy 82-111 criteria by the ERF Appraisal Program, items III.A.2.4, III.A.2.5, III.A.2.6, and III.A.2.8 are considered closed because the criteria specific to NUREG-0737 no longer fully apply.

No items of noncompliance or deviations were identified.

14. Regulatory Performance Improvement Plan

Commonwealth Edison Company implemented a Regulatory Performance Improvement Plan (RPIP) in February 1984. The plan concept was a formal effort to improve safety and error-free operations. This was developed in response

to NRC concerns over recent errors and escalated enforcement actions. During the inspection period, the inspectors have observed licensee actions such as followup on potentially significant events, conduct of operations, cleaning, painting and improvements in appearance of the plant. In addition, the inspectors have had discussions with shift overview superintendents (SOS), the station superintendent, and various corporate personnel. It appears that there is an improving trend. However, further observations are necessary to form a conclusion. This matter will be observed and addressed in future inspections.

No items of noncompliance or deviations were identified in this area.

15. Unit 1 Chemical Cleaning

The inspectors observed the licensee's preparation for the chemical cleaning of Unit 1 currently planned for July 1984. The inspectors reviewed new and modified procedures, interviewed personnel involved in the project, and toured the facilities. This will continue to be observed as part of the routine inspection program until the project is complete.

No items of noncompliance or deviations were identified in this area.

16. Independent Inspection

Unit 3, Main Turbine Repair

During the inspection period, the inspectors followed the repair activities on the Unit 3 high pressure turbine and verified that adequate radiation protection precautions were being implemented.

No items of noncompliance or deviations were identified in this area.

17. Report Review

During the inspection period, the inspectors reviewed the licensee's Monthly Operating Reports for March and April 1984. The inspectors confirmed that the information provided met the requirements of Technical Specification 6.6.A.3 and Regulatory Guide 1.16.

No items of noncompliance or deviations were identified in this area.

18. Meeting with Local Municipal Officials

On May 10, 1984, at 7:00 pm, a meeting was held for local public officials in the board meeting room of the Grundy County Court House. The purpose of the meeting was to provide an opportunity for local public officials to meet the resident inspectors and associated Region III personnel, and discuss to the Resident Inspection Program for Dresden. The meeting was attended by approximately 26 officials, their guests, and several individuals from State of Illinois agencies. The major areas of interest were

in emergency preparedness and its demand on local resources with minimal compensation, the resident inspectors' roles in daily plant activities, and the NRC enforcement program. At the conclusion, the Grundy County Sheriff provided a tour of the emergency preparedness communications and notification facilities for the NRC personnel.

19. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, items of noncompliance, or deviations. Unresolved items addressed during the inspection are discussed in Paragraph 2.

20. Open Items

Open items are matters which have been discussed with the licensee, which will be reviewed further by the inspector, and which involve some action on the part of the NRC or licensee or both. Open items addressed during the inspection are discussed in Paragraph 2.

21. Exit Interview

The inspectors met with lieensee representatives (denoted in Paragraph 1) throughout the inspection period and at the conclusion of the inspection on May 21, 1984, and summarized the scope and findings of the inspection activities. The licensee acknowledged the findings of the inspection.

Tab 2

DRESDEN 2 & 3

FIRE PROTECTION PROGRAM DOCUMENTATION PACKAGE

Inspection Report No. 50-010/84-09, 50-237/84-11, 50-249/84-10

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

JUL 3 1984

JUL 6 RECD

Docket No. 50-10
Docket No. 50-237
Docket No. 50-249

Commonwealth Edison Company
ATTN: Mr. Cordell Reed
Vice President
Post Office Box 767
Chicago, IL 60690

Gentlemen:

This refers to the routine safety inspection conducted by Messrs. T. M. Tongue and S. Stasek of this office during the period of May 22 through June 15, 1984 of activities at Dresden Nuclear Power Station, Units 1, 2 and 3, authorized by NRC Operating Licenses No. DPR-02, DPR-19, and DPR-25, and to the discussion of our findings with Mr. D. Scott and others of your staff at the conclusion of the inspection.

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.

During this inspection, certain of your activities appeared to be in noncompliance with NRC requirements, as specified in the enclosed Appendix. A written response is required.

In accordance with 10 CFR 2.790(a), a copy of this letter and the enclosure(s) will be placed in the NRC Public Document Room unless you notify this office, by telephone, within ten days of the date of this letter and submit written application to withhold information contained therein within thirty days of the date of this letter. Such application must be consistent with the requirements of 2.790(b)(1). If we do not hear from you in this regard within the specified periods noted above, a copy of this letter, the enclosure(s), and your response to this letter will be placed in the Public Document Room.

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

JUL 3 1984

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

Sincerely,

W D Shafer

W. D. Shafer, Chief
Reactor Porjects Branch 2

Enclosures:
1. Appendix, Notice of Violation
2. Inspection Report No. 50-010/84-09(DRP); No. 50-237/84-11(DRP); No. 50-249/84-10(DRP)

cc w/encl:
D. L. Farrar, Director of Nuclear Licensing
D. J. Scott, Station Superintendent
DMB/Document Control Desk (RIDS)
Resident Inspector, RIII
Phyllis Dunton, Attorney General's Office, Environmental Control Division

Appendix

NOTICE OF VIOLATION

Commonwealth Edison Company

Docket No. 50-237
Docket No. 50-249

As a result of the inspection conducted on May 22 through June 15, 1984, and in accordance with the General Policy and Procedures for NRC Enforcement Actions, (10 CFR Part 2, Appendix C), the following violation was identified:

Dresden Technical Specification 4.12.D.3 states in part "At least once per operating cycle, the (Cardox) system valves and associated dampers will be verified to actuate automatically and manually."

Contrary to the above, the Cardox system discharge master valve was not tested to verify actuation in the automatic mode and it appears that testing in the automatic mode was not included in the surveillance program.

Pursuant to the provisions of 10 CFR 2.201, you are required to submit to this office within thirty days of the date of this Notice a written statement or explanation in reply, including for each item of noncompliance: (1) corrective action taken and the results achieved; (2) corrective action to be taken to avoid further noncompliance; and (3) the date when full compliance will be achieved. Consideration may be given to extending your response time for good cause shown.

JUL 3 1984
Dated

WDShafer
W. D. Shafer, Chief
Reactor Projects Branch 2

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-010/84-09(DPRP); 50-237/84-11(DPRP); 50-249/84-10(DPRP)

Docket Nos. 50-010; 50-237; 50-249 Licenses No. DPR-02; DPR-19; DPR-25

Licensee: Commonwealth Edison Company
P. O. Box 767
Chicago, IL 60690

Facility Name: Dresden Nuclear Power Station, Units 1, 2, and 3

Inspection At: Dresden Site, Morris, IL

Inspection Conducted: May 22 through June 15, 1984

Inspectors: T. M. Tongue

S. Stasek

Approved By: N. J. Chrissotimos, Chief
Reactor Projects Branch 2C

6-27-84
Date

Inspection Summary

Inspection during the period of May 22 through June 15, 1984 (Report Nos. 50-10/84-09(DPRP); 50-237/84-11(DPRP); 50-249/84-10(DPRP))
Areas Inspected: Routine unannounced resident inspection of action on previous inspection findings, headquarters request, regional request, operational safety, fire protection program, surveillance program, maintenance, licensee event reports, I.E. Information Notices, Unit 1 chemical cleaning, spent fuel shipments, and report review. The inspection involved a total of 122 inspector-hours onsite by 2 NRC inspectors including 22 inspector-hours onsite during off-shifts.
Results: Of the 12 areas inspected no items of noncompliance or deviations were identified in 11 areas; one item of noncompliance was identified in one area (inadequate surveillance testing of cardox system - paragraph 6).

DETAILS

1. Persons Contacted

*D. Scott, Station Superintendent
R. Ragan, Operations Assistant Superintendent
J. Eenigenburg, Maintenance Assistant Superintendent
J. Wujciga, Administrative and Support Services Assistant Superintendent
J. Brunner, Technical Staff Supervisor
R. Christensen, Unit 1 Operating Engineer
J. Almer, Unit 2 Operating Engineer
T. Ciesla, Unit 3 Operating Engineer
D. Sharper, Waste Systems Engineer
G. Myrick, Radiation Chemistry Supervisor
B. Saunders, Station Security Administrator
M. Dillon, Station Fire Marshall
S. McDonald, Radiation Protection Supervisor
J. Achterberg, Assistant Technical Staff Supervisor
D. Ringo, Surveillance Coordinator
*R. Stobert, Quality Assurance Inspector

Contractor:

K. Jones, Driver, Home Transportation Corporation

State of Illinois:

V. Muzzallupo, Illinois Department of Nuclear Safety
R. Reese, Hazardous Materials Officer, Illinois State Police

The inspector also talked with and interviewed several other licensee employees, including members of the technical and engineering staffs, reactor and auxiliary operators, shift engineers and foremen, electrical, mechanical and instrument personnel, and contract security personnel.

*Denotes those attending one or more exit interviews conducted on May 25 and June 15, 1984, and informally at various times throughout the inspection period.

2. Action on Previous Inspection Findings

(Closed) Open item (237/79-20-01(DPRP)): Loss of secondary containment integrity. The licensee redesigned the blow-off bolts for the panels and has permanently blocked the supply fan vortex dampers to assure proper containment integrity.

(Closed) Unresolved item (237/83-21-02(DPRP)): Dresden Technical Specifications do not adequately address the limits on allowable primary containment leakage during plant operation. By memo dated April 4, 1984, an interpretation was made by NRR on this matter which essentially agreed with the licensee's interpretation. That is, La is considered to be the leakage limit during plant operation and 0.75 La is limiting only during the performance of a Type A test.

No further items of noncompliance or deviations were identified.

3. Headquarters Request

By memorandum dated May 24, 1984, the Director of the Office for Analysis and Evaluation of Operational Data requested the resident inspectors to review certain occurrences with the licensee for reportability under 10 CFR 50.73. The events in question were reported through the NRC's "Morning report" system but were not reported as licensee events (LERs) pursuant to 10 CFR 50.73. The specific event at Dresden was when the licensee reported, per 10 CFR 50.72, "Immediate Notifications" of being in an unusual event due to outages on redundant ECCS equipment. Further evaluation revealed that personnel on earlier shifts had considered the circumstances and that the situation was permitted by Technical Specifications. The resident inspectors discussed this with the licensee at the time and agreed that the licensee had not been in an unusual event. The morning report was written as a followup to the ENS phone call. The event occurred on January 26, 1984. No further action is considered necessary on this issue.

No items of noncompliance or deviations were identified in this area.

4. Regional Request

A regional request was reviewed by the resident inspectors for applicability at Dresden based on an event identified at the Quad Cities (QC) nuclear station. (Reference inspection report 50-254/83-04(DPRP); 50-265/84-03(DPRP))

During a 125 volt D.C. battery discharge test at QC Unit 1, it was found that the discharge rate was at 85 amperes steady state. The Final Safety Analysis Report (FSAR) stated that the battery discharge rate should be 62.3 amperes for 8 hours.

Investigation revealed that modifications (additional loads) have been added to the battery and it appears that the added loads were not reviewed pursuant to 10 CFR 50.59 for its effect on the battery capability. Subsequent evaluation revealed that the battery, under present conditions, would have insufficient capacity under certain accident conditions.

The 10 CFR 50.59 reviews were conducted within the Station Nuclear Engineering Department (SNED) for both QC and Dresden and it appears that the same omission occurred for Dresden. However, review of records at Dresden since 1981 show no evidence of exceeding a level of 60 amperes (nominally loads were about 52 amps). In addition, unlike QC, the Dresden FSAR shows a specific list of battery loads. Personnel at Dresden are also submitting an Action Item to SNED for a battery load profile review.

The information on the QC battery has been submitted to NRR for review and evaluation. The outcome of that review will be used to determine enforcement action at QC as well as Dresden. This issue is presently considered an unresolved inspection item. (50-237/84-11-01(DPRP); 50-249/84-10-01(DPRP)).

No items of noncompliance or deviations were identified in this area.

5. Operational Safety Verification

The inspectors observed control room operations, reviewed applicable logs and conducted discussions with control room operators during the period from May 22 to June 15, 1984. The inspectors verified the operability of selected emergency systems, reviewed tagout records and verified proper return to service of affected components. Tours of Unit 2 reactor building and turbine building were conducted to observe plant equipment conditions, including potential fire hazards, fluid leaks, and excessive vibrations and to verify that maintenance requests had been initiated for equipment in need of maintenance.

During the inspection period while Unit 3 was in an outage to repair the main turbine, the inspectors verified that surveillance tests were conducted, containment integrity requirements were met, and emergency systems were available as necessary.

Throughout the entire inspection period, Unit 1 remained in a longterm shutdown condition with all fuel removed from the vessel. The inspectors verified that all applicable requirements for Unit 1 were met during this period.

The inspectors, by observation and direct interview, verified that the physical security plan was being implemented in accordance with the station security plan.

The inspectors observed plant housekeeping/cleanliness conditions and verified implementation of radiation protection controls. During the inspection, the inspectors walked down the accessible portions of the following systems to verify operability by comparing system lineup with plant drawings, as-built configuration or present valve lineup lists; observing equipment conditions that could degrade performance; and verified that instrumentation was properly valved, functioning, and calibrated.

a. Unit 2

Low Pressure Coolant Injection System (both loops), Core Spray System (both loops), and Isolation Condenser.

b. Unit 3

Low Pressure Coolant Injection System ("B" loop), Core Spray System ("B" loop), Isolation Condenser, and Unit 3 Emergency Diesel Generator.

c. Unit 2/3 (Common)

Standby Gas Treatment System, Cardox System, Halon System, and Fire Water System.

The inspectors reviewed new procedures and changes to procedures that were implemented during the inspection period. The review consisted of a verification for accuracy, correctness, and compliance with regulatory requirements. The inspectors also witnessed portions of the radioactive waste system controls associated with radwaste shipments and barreling.

These reviews and observations were conducted to verify that facility operations were in conformance with the requirements established under technical specifications, 10 CFR, and administrative procedures.

No items of noncompliance or deviations were identified in this area.

6. Fire Protection Program

During the inspection period, the inspector reviewed the licensee's fire protection program against Technical Specification requirements and licensee commitments. The inspector verified that welding and cutting operations along with other activities involving open flame ignition sources in safety related areas were properly performed in conformance with appropriate station procedures. Walkdowns were conducted of the accessible portions of the cardox system, halon system, water suppression system, and portable fire protection equipment. Proper housekeeping in safety related areas was also verified during plant tours. Training sessions and periodic drills for fire brigade members were found to be inplace and acceptable. The following surveillances were reviewed for adequacy and completeness:

DFPP 4123-5 "Diesel Fire Pump Week Operability"
DFPP 4132-1 "Verification of U-2/3 Sprinkler Systems Integrity"
DFPP 4132-2 "Verification of Unit 2 Sprinkler Systems Integrity"
DFPP 4132-3 "Verification of Unit 3 Sprinkler Systems Integrity"
DFPP 4145-1 "Cardox System Semi-Annual Maintenance Test"
DFPP 4153-2 "Emergency Lighting Monthly Inspection"
DFPP 4175-1 "Fire Stop Integrity and Maintenance"
DFPP 4175-2 "Operating Fire Stop/Barrier Surveillance"
DFPP 4175-3 "Shutdown Fire Stop Surveillance"
DFPP 4185-3 "Fire Detection System Operation"
DFPP 4195-1 "Halon Systems Semi-Annual Maintenance"

While reviewing surveillance DFPP 4145-1, the inspector noted that the master discharge valve, in the Cardox system was not verified to open on an automatic initiation of the system. This valve is located on the discharge piping downstream of the CO_2 storage tank, and is used to pressurize the system's main header in the event that a fire is sensed in any one of the areas protected by the Cardox system (including all three emergency diesel generator rooms). When station management was made aware of the deficiency in the surveillance, the Cardox system was declared inoperable and a special procedure (SP 84-5-35) was written to test the master discharge valve. The valve was subsequently tested the same day and verified to operate correctly. The inspector reviewed the procedure and witnessed the test and found all aspects of the surveillance to be acceptable. Following the successful completion of the test, the licensee again declared the system operable.

Dresden Technical Specification 4.12.D.3 requires that all valves in the Cardox system be tested at least once per operating cycle to verify each will actuate manually and automatically. Because the master discharge valve was not tested for automatic actuation in accordance with the aforementioned requirement, this is considered an item of noncompliance (237/84-11-02(DRP); 249/84-10-02(DRP)).

One item of noncompliance was identified in this area.

7. <u>Surveillance Observation</u>

The inspectors observed Technical Specifications required surveillance testing on the Unit 2/3 (Swing) Emergency Diesel Generator and verified that testing was performed in accordance with adequate procedures, that test instrumentation was calibrated, that limiting conditions for operation were met, that removal and restoration of the affected components were accomplished, that test results conformed with Technical Specifications and procedure requirements and were reviewed by personnel other than the individual directing the test, and that any deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel.

The inspectors also witnessed/reviewed portions of the following test activities:

<u>Unit 2</u>

Core Spray System Pump Test With Torus Available, Core Spray System Valve Operability Check, Low Pressure Coolant Injection (LPCI) System Valve Operability Test, and LPCI System Pump Operability Test With Torus Available.

<u>Unit 3</u>

Daily/Weekly Storage Battery Check.

<u>Unit 2/3 (Common)</u>

Cardox System Master Valve Operability Test.

The inspector also reviewed the master surveillance program for testing and calibration as required by technical specifications. This involved a verification of frequencies, responsible plant groups and test status. The inspector tested the system to verify that recent technical specifications had been appropriately addressed and, that formal methods and responsibilities had been defined for review of test data and reporting deficiencies, etc.

No items of noncompliance or deviations were identified in this area.

8. Maintenance Observation

Station maintenance activities of safety related systems and components listed below were observed/reviewed to ascertain that they were conducted in accordance with approved procedures, regulatory guides and industry codes or standards and in conformance with Technical Specifications.

The following items were considered during this review: the limiting conditions for operation were met while components or systems were removed from service; approvals were obtained prior to initiating the work; activities were accomplished using approved procedures and were inspected as applicable; functional testing and/or calibrations were performed prior to returning components or systems to service; quality control records were maintained; activities were accomplished by qualified personnel; parts and materials used were properly certified; radiological controls were implemented; and, fire prevention controls were implemented.

Work requests were reviewed to determine status of outstanding jobs and to assure that priority is assigned to safety related equipment maintenance which may affect system performance.

The following maintenance activity was observed/reviewed:

Unit 3

Emergency Diesel Generator Bi-Annual Maintenance

No items of noncompliance or deviations were identified in this area.

9. Licensee Event Reports Followup

Through direct observations, discussions with licensee personnel, and review of records, the following event reports were reviewed to determine that reportability requirements were fulfilled, immediate corrective action was accomplished, and corrective action to prevent recurrence had been accomplished in accordance with technical specifications.

Unit 2

(Closed)	237/82-49	Torus Inert and Purge Valve 2-1601-56 Did Not Auto-Close During Surveillance.
(Closed)	237/83-08	Indications Discovered During Inservice Inspection.
(Closed)	237/83-12	Mechanical Snubber Failure on Main Steam Line.
(Closed)	237/83-29	Excessive Leakage Found During Integrated Leak Rate Test.

Unit 3

(Closed)	249/82-06	Missed Battery Quarterly Surveillance.
(Closed)	249/82-14	Discovery of Crack in Unit 3 Head Seal Leak-Off Line.
(Closed)	249/84-04	Reactor Scram While Performing Surveillance.

The preceding LERs have been reviewed against the criteria of 10 CFR 2, Appendix C, and when the incidents described meet all of the following requirements, no Notice of Violation is normally issued for that item.

a. The event was identified by the licensee,
b. The event was an incident that, according to the current enforcement policy, met the criteria for severity levels IV or V violations,
c. The event was appropriately reported,
d. The event was or will be corrected (including measures to prevent recurrence within a reasonable amount of time), and
e. The event was not a violation that could have been prevented by the licensee's corrective actions for a previous violation.

No items of noncompliance or deviations were identified in this area.

10. I.E. Information Notice Followup

Each of the following I.E. Information Notices (IEN) was reviewed by the Resident Inspector to verify 1) that the information notice was received by licensee management, 2) that a review for applicability was performed, and 3) that if the information notice was applicable to the facility, appropriate actions were taken or were scheduled to be taken.

(Closed) IEN 83-01: Ray Miller, Inc. I.E. Bulletin subsequently issued addressing this matter.

(Closed) IEN 83-02: Limitorque HOBC, H1BC, H2BC, and H3BC Gearheads. Not applicable to Dresden.

(Closed) IEN 83-03: Calibration of Liquid Level Instruments. Density and temperature compensation is accounted for during calibration operations.

(Closed) IEN 83-04: Failure of ELMA Power Supply Units. Not applicable to Dresden.

16. Open Items

Open items are matters which have been discussed with the licensee, which will be reviewed further by the inspector, and which involve some action on the part of the NRC or licensee or both. An open item addressed during the inspection is discussed in Paragraph 2.

17. Exit Interview

The inspectors met with licensee representatives (denoted in Paragraph 1) throughout the inspection period and at the conclusion of the inspection on June 15, 1984, and summarized the scope and findings of the inspection activities. The licensee acknowledged the findings of the inspection.

Commonwealth Edison
One First National Plaza Chicago Illinois
Address Reply to Post Office Box 767
Chicago, Illinois 60690

July 30, 1984

Mr. James G. Keppler
Regional Administrator
U.S. Nuclear Regulatory Commission
Region III
799 Roosevelt Road
Glen Ellyn, IL 60137

Subject: Dresden Station Units 1, 2, and 3
Response to Inspection Report Nos.
50-010/84-09, 50-237/84-11 and
50-249/84-10
NRC Docket Nos. 50-010, 50-237 & 50-249

Reference (a): W. D. Shafer letter to Cordell Reed
dated July 3, 1984.

Dear Mr. Keppler:

This letter is in response to the inspection conducted by Messrs. T. M. Tongue and S. Stasek during the period of May 22 thru June 15, 1984, of activities at Dresden Station. Reference (a) indicated that certain activities appeared to be in noncompliance with NRC requirements. The Commonwealth Edison Company response to the Notice of Violation is provided in the enclosure.

If you have any further questions on this matter, please direct them to this office.

Very truly yours,

D. L. Farrar
Director of Nuclear Licensing

lm

Attachment

cc: NRC Resident Inspector - Dresden

8996N

ATTACHMENT A

COMMONWEALTH EDISON COMPANY

RESPONSE TO NOTICE OF VIOLATION

Dresden Technical Specification 4.12.D.3 states in part:

> "At least once per operating cycle, the (Cardox) system valves and associated dampers will be verified to actuate automatically and manually."

Contrary to the above, the Cardox system discharge master valve was not tested to verify actuation in the automatic mode and it appears that testing in the automatic mode was not included in the surveillance program.

DISCUSSION

During a routine NRC inspection from May 22 through June 15, 1984 of Dresden's Fire Protection Program, it was discovered that no documentation existed to verify that the Cardox system electro-mechanical master pilot valve was operable in the automatic mode. This master pilot valve controls the position of the selector valves which control the flow of CO_2 into each of the diesel generator rooms. In reviewing Procedure DFPP 4145-1, Cardox System Semi-Annual Maintenance Test, Revision 1, it was found that the master pilot valve was tested only for manual actuation i.e., the procedure did not address a test for verifying automatic operation. Since the surveillance interval outlined in Technical specification 4.12.D.3 was exceeded, the Cardox system was immediately declared inoperable and an hourly fire inspection was established per Technical Specification 3.12.D.4. Also, backup fire suppression equipment was provided for these areas. A Special Procedure was written, on-site reviewed and performed to verify automatic actuation of the master pilot valve. Upon completion of the Special Procedure the Cardox system was returned to service and the hourly fire inspection was discontinued.

CORRECTIVE ACTION TAKEN TO AVOID FURTHER NONCOMPLIANCE

The corrective action taken to avoid further non-compliance was to incorporate the Special Procedure for testing the automatic function of the master pilot valve into the existing Cardox System Semi-Annual Maintenance Test, DFPP 4145-1.

Also, a review of Dresden's Technical Specification Section 4.12 has been initiated to verify that all surveillance items are performed within their specified time intervals using approved station procedures.

DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED

Procedure DFPP 4145-1 will be revised by August 31, 1984 and the Technical Specification review will be completed by September 28, 1984.

8996N

Tab 3

DRESDEN 2 & 3

FIRE PROTECTION PROGRAM DOCUMENTATION PACKAGE

Inspection Report No. 50-237/85033, 50-249/85029

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

NOV 14 1985

Docket No. 50-237 85-33
Docket No. 50-249 85-29

Commonwealth Edison Company
ATTN: Mr. Cordell Reed
Vice President
Post Office Box 767
Chicago, IL 60690

Gentlemen:

This refers to the routine safety inspection conducted by Messrs. J. Holmes and C. Ramsey of this office on September 30 through October 21, 1985, of activities at Dresden Nuclear Power Station, Units 2 and 3, authorized by NRC Operating Licenses No. DPR-19 and No. DPR-25 and to the discussion of our findings with Mr. D. Scott at the conclusion of the inspection.

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.

During this inspection, certain of your activities appeared to be in violation with NRC requirements. These issues, identified in paragraphs 3 and 7.a of the enclosed inspection report, are being reviewed for potential escalated enforcement action. The results of that review will be forwarded to you by separate correspondence which will identify the nature of expected formal response.

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

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

NOV 14 1985

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

Sincerely,

Carl J. Papeiello

Carl J. Papeiello, Director
Division of Reactor Safety

Enclosure: Inspection Reports
No. 50-237/85033(DRS); and
No. 50-249/85029(DRS)

cc w/enclosure:
D. L. Farrar, Director
of Nuclear Licensing
D. J. Scott, Plant Manager
DCS/RSB (RIDS)
Licensing Fee Management Branch
Resident Inspector, RIII
Phyllis Dunton, Attorney
General's Office, Environmental
Control Division

RIII	RIII	RIII	RIII	RIII	RIII
Ramsey	Holmes	Guldemond	Wright	Reyes	Papeiello
11/14/85	11/14/85	11/14/85	11/14/85	11/14/85	11/14/85

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-237/85033(DRS); 50-249/85029(DRS)

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

Licensee: Commonwealth Edison Company
P. O. Box 767
Chicago, IL 60690

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

Inspection At: Morris, IL

Inspection Conducted: September 30 through October 21, 1985

Inspectors: J. Holmes 11-13-85 Date

C. Ramsey 11-13-85 Date

Approved By: W. G. Guldemond, Chief 11-13-85 Date
Operational Programs Section

Inspection Summary

Inspection on September 30 through October 21, 1985 (Reports No. 50-237/85033(DRS); 50-249/85029(DRS))
Areas Inspected: Routine, unannounced safety inspection conducted to verify the adequacy of the facility's fire protection program implementation and to determine the status of LERs and previous open items. The inspection involved 71 inspector-hours by two NRC inspectors including 2 inspector-hours onsite during off-shifts and 11 inspector-hours conducting in-office review at the Region III office.
Results: Of the 6 areas inspected, no violations or deviations were identified in four areas. Two violations were identified in the remaining two areas (failure to adhere to program staffing requirements - Paragraph 3; failure to comply with a license condition to install an automatic fire detection system in the Reactor Building refueling floor area - Paragraph 7a).

DETAILS

1. Persons Contacted

DNPS

D. Adam, Compliance Administrator
*J. Brunner, Assistant Superintendent, Technical Services
T. Ciesla, Assistant Superintendent, Operations
*M. Dillon, Fire Marshall
*R. Flissner, Service Superintendent
*T. Hausheer, Nuclear Services, Technical
*P. Lau, QA Supervisor
*J. McDonald, Station Nuclear Engineering
*B. Rybak, Station Nuclear Engineering
*D. Scott, Station Manager
*R. Whalen, Technical Staff
J. Wujciga, Production Superintendent

US NRC

E. Hare, Resident Inspector
*L. McGregor, Senior Resident Inspector
S. Stasek, Resident Inspector

*Denotes those in attendance at the exit meeting of October 4, 1985.

2. Licensee Actions on Previous Inspection Findings

a. (Open) LER (237/85029) and Violation (237/85028-01): Auxiliary electric equipment room halon system declared inoperable due to ventilation dampers failing to close. Fire watch was not established per Technical Specification No. 3.12.H.2.

Region III's followup of this event is documented in Inspection Report No. 50-237/85028(DRS). As a result of this followup violation No. 237/85028-01 was issued. No response to this violation was required because the licensee's interim and long term corrective actions were determined satisfactory. The interim corrective actions were implemented prior to or during the followup inspection. The proposed long term corrective actions have not been implemented. Therefore, this event report remains open.

b. (Closed) LER (249/85014): Wet pipe sprinkler system in Unit 3 turbine trackway had to be rerouted to allow for overhead clearance for new turbine rotors. The sprinkler system was out of service 5 1/2 hours beyond the 14 day limit permitted by Technical Specification 3.12.C.3.

The event report is closed based on the licensee's corrective actions taken which included restoration of the system to service and functional testing prior to declaring the system operable.

c. (Closed) LER (237/85010): Fire door for the Unit 2 125V DC battery room found open. A fire watch was not established within one hour per Technical Specification 3.12.F.2.

This event report is closed based on the licensee's corrective actions taken which included immediate closure of the fire door and training/instruction of plant operators on the requirements to keep fire doors closed at all times when not in use.

d. (Open) LER (237/84-20): Two of seven root valves that were installed on fire hose stations were found to be in the closed position rendering the fire hose stations inoperable.

Although the licensee's corrective actions for this event included prompt opening of the closed root valves, the inspectors determined that the licensee's program for administratively controlling valves that are not electrically supervised using wire seals to secure these valves in the open position and performing monthly inspections to verify valve positions does not appear to be working. During the inspection the inspectors observed several non-electrically supervised valves in the fire protection system with missing or damaged wire seals.

To correct this problem, the licensee stated that the program for administratively controlling these valves is being upgraded to include locking these valves (chain and lock) in the open position in addition to monthly inspections to verify each valve position in accordance with NFPA Standard 26. This event report remains open pending Region III verification of the licensee's upgraded corrective actions.

e. (Open) LER (237/84-17; 237/84-05): Failure to establish continuous or hourly fire watch patrols due to inoperability of all or portions of fire detection and sprinkler alarm systems in the control room.

During these events and at the present time, the fire detection and sprinkler alarm system printer indications are interlocked (dependent) into the plant security system computer. Indication of fire detection and sprinkler alarms in the control room was lost in two events either because of a loss of power to the plant security system computer or because of modifications being made to upgrade the plant security system computer. Apparently, any failure of the plant security system computer can cause the loss of all or portions of fire detection and sprinkler alarm annunciation in the control room.

This installation does not comply with the licensee's commitment to NFPA 72D as stated in the licensee's April 1977 response (point-by-point comparison) to Appendix A to NRC Branch Technical Position

(BTP) APCSB 9.5-1. The licensee's scheduled plant modification No. M12-2/3-84-109 identifies corrective action for this problem as the installation of independent circuits for fire detection and sprinkler system alarms which alarm and annunciate in the control room in accordance with NFPA 72D. This modification is scheduled to be completed in December 1985. These LERs will remain open pending Region III verification of the licensee's corrective actions.

f. (Closed) LER (237/84-11): Fire wall penetrations to Unit 2/3 diesel generator rooms were not sealed.

This event report is closed based on the licensee's corrective actions taken, which include establishment of a fire watch within one hour per Technical Specification No. 3.12.F.2 and sealing the penetrations per drawing No. 12E-6058.

g. (Closed) LER (237/84-08): NRC inspection of the licensee's compliance with fire protection Technical Specification surveillance requirements identified that a cardox system master valve was not being tested in the automatic mode. The master valve test procedure was written to test the valve in the manual mode.

This event report is closed based on the licensee's corrective actions taken which included prompt removal of the master valve from service, revision of the surveillance test procedure, and satisfactory testing of the valve in the automatic mode.

h. (Closed) LER (249/83-34/03L): Unit 3 trackway sprinkler system out of service due to damage by mobile crane boom.

This event is closed based on the licensee's corrective actions taken, which included making the necessary system repairs, prompt restoration of the system to service and instructions to plant personnel regarding the movement of mobile cranes and the fragility of systems and components in their path.

i. (Closed) LER (249/83-17/03L): HPCI deluge system solenoid valve taken out of service because the valve would not reset.

This event report is closed based on the licensee's corrective actions taken, which included prompt removal of the deluge system from service, making the necessary repairs to the solenoid valve and restoration of the system to service.

j. (Closed) LER (237/81-15/03L): Unit 2/3 diesel generator room CO_2 system heat detector surveillances not performed per Technical Specification 4.12.A.1.

This event report is closed based on the licensee's corrective actions taken, which included instruction to plant personnel to perform the required heat detector surveillances and satisfactory performance of the surveillance.

k. (Closed) Violations (237/81-09-01; 249/81-06-01): Four penetration seals identified as being defective were inoperable for an excessive period of time. Neither prompt nor timely corrective action was taken.

This item is closed based on the licensee's June 29, 1981 response to Region III which discussed the licensee's corrective actions taken to avoid future violations in this area. The inspector's review of Procedure No. DFPP-4175-2, Revision 4, indicated that appropriate instructions are provided to plant personnel which refer to detail drawings for proper installation of penetration fire seals.

l. (Open) Violations (237/81-09-03; 249/81-06-03): (a) Fifty percent of fire extinguishers sampled did not have 1981 monthly inspection tags attached; (b) 5 year hydrostatic test for portable CO_2 extinguisher cylinders were overdue; and (c) numerous compressed gas cylinders were improperly stored.

The licensee's corrective actions identified in their June 29, 1981 response to items (a) and (b) of this violation were ineffective. Subsequent QA audits and surveillance by the licensee's onsite QA department have revealed that these deficiencies are continuing. For example, deficiencies identified in QA surveillance No. QAS 12-85-236 for the period September 23 through 29, 1985 include the following: wrong date on extinguisher tags; extinguishers past due for 5 year hydro testing; no service date on extinguisher tags; no seal on extinguisher pull pin; partially discharged extinguisher.

During plant tours by the inspectors, identical deficiencies were observed. In one instance, a CO_2 portable extinguisher hose was damaged to the extent that the webbing in the hose was exposed. A hole existed in the webbing that may have allowed the extinguishing agent to escape through the hose prior to reaching the CO_2 discharge nozzle. This extinguisher was located on fire cart No. 2. In addition, the inspectors observed that wheeled dry chemical extinguishing units Nos. PK 21 and PK 22 had tags which indicated that surveillances were missed the months of May and September 1985. The continuing existence of this type of deficiency is indicative of a lack of management attention in this area. Management attention and staffing is the subject of a violation documented in paragraph 3 of this report. Your response to that violation should address your corrective actions for failing to properly maintain fire extinguishers. These items will remain open pending the further review of the licensee's corrective actions by Region III.

Item (c) of this violation is closed based on the licensee's corrective action taken which included the installation of metal storage racks for compressed gas cylinders, securing the cylinders with metal chains and revision of Procedure No. DAP 3-11.

m. (Open) Unresolved Item (237/81-09-06; 249/81-06-06): Fire brigade drills and training do not appear to meet the intent of NRC requirements.

Section 6.0 of the original fire protection SER, dated March 1978, recommended that the licensee's administrative controls follow the guidelines set forth in the NRC Guidance Document entitled "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance." A supplement to the original SER was issued December 2, 1980. Section 3.1 of this supplemental SER closes out the issue of administrative controls with the NRC staff's acceptance of the licensee's discussion of administrative controls provided in letters dated January 24, February 24, March 20 and July 27, 1978, January 31, and April 30, 1979. Therefore, Section III.1.3.b of Appendix R is not applicable to administrative controls for fire protection at Dresden.

Based on the licensee's submittals discussed above, the NRC staff concluded that the licensee's administrative controls for fire protection met NRC guidelines and, the applicable regulatory requirement for fire protection administrative controls at Dresden is the Commission's guidance issued on the implementation of General Design Criterion 3 of Appendix A to 10 CFR Part 50 for existing power plants.

Section 2.0 and 3.0 of Attachment No. 2 to NRC Guidance Document "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance" requires practice sessions be held for fire brigade members to provide each brigade member with experience in actual fire extinguishment and the use of emergency breathing apparatus under strenuous conditions. Fire brigade drills are required to be performed so that the fire brigade can practice as a team. The drills are to be performed at regular intervals but not to exceed three months for each fire brigade. The drills are required to be critiqued to assess each brigade member's knowledge of his role in fire fighting strategy.

The licensee is not meeting these requirements for the following reasons:

(1) By attempting to meet the requirements contained in Section III.1.3.b of Appendix R to 10 CFR 50, the licensee has been conducting one fire drill per month with the intent of getting all designated fire brigade members involved in at least two drills per year.

(2) Practice sessions that provide each brigade member with actual fire extinguishment experience and the use of emergency breathing apparatus under strenuous conditions (full fire fighting gear) have not been conducted due to a breakdown in contractual arrangements with an independent firm.

(3) Fire brigade drills have not been critiqued at three year intervals by qualified individuals independent of the licensee's staff.

To resolve this concern, the licensee is requested to make available a detailed assessment of fire brigade drills, practice sessions and three year audits of fire brigade drills by qualified individuals independent of the licensee's staff. This assessment should establish whether the licensee is in compliance with commitments made to the NRC which resulted in the NRC staff conclusions that the licensee's administrative controls for fire protection were acceptable.

This item remains open pending region review of the licensee's assessment.

n. (Open) Unresolved Item (237/81-09-07; 249/81-06-07): Specific pre-fire fighting plans or strategies for all safety-related areas and areas presenting a hazard to safety-related equipment were not developed and implemented.

As discussed in item 237/81-09-06; 249/81-06-06 above, the requirements of 10 CFR 50, Appendix R (Sections III.K.11 and 12) are not applicable in this case. The applicable requirements are contained in Attachment No. 5 of NRC Guidance Document "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance."

During the inspection, the licensee provided the inspectors with a copy of pre-fire plans that contained specific fire fighting strategies for fighting fires in all safety-related areas and areas that present a hazard to safety-related equipment. The pre-fire plans appear to provide adequate fire fighting procedures and instructions. However, these plans have not been implemented. According to the licensee, the plans will be implemented and incorporated into fire brigade training lesson plans by the end of the first quarter of 1986. This item will remain open pending said implementation.

o. (Open) Unresolved Item (237/81-09-08; 249/81-06-08): Fire brigade practice sessions have not been conducted in accordance with commitments made to the NRC. A hands-on practice session was held in 1979 with full brigade attendance, but no practice session was held in 1980.

This item will remain open pending Region III review of the licensee's response to Item b of Unresolved Item No. 237/81-09-06; 249/81-06-06 as discussed in this report.

3. <u>Fire Protection Program Organization and Personnel Staffing</u>

10 CFR 50.48 requires that each operating nuclear power plant have a fire protection plan that satisfies Criterion 3 of Appendix A to 10 CFR 50. Except for the requirements of Section III.G, III.J, and III.O of Appendix R to 10 CFR 50, the approved fire protection plan that satisfies Criterion 3 of Appendix A to 10 CFR 50 is discussed in the original fire protection SER, dated March 1978, a fire protection SER Supplement, dated December 2, 1980, and the licensee's Fire Hazard Analysis submittals entitled

"Information Relevant to Fire Protection Systems and Programs" dated October 1976, January 1977, and April 1977. Furthermore, the licensee committed to follow certain NRC Supplemental Guidance Documents as discussed in letters to the NRC, dated January 24, February 24, March 20 and July 27, 1978; January 31 and April 30, 1979.

The requirements for overall responsibility for the Fire Protection Program are discussed in Sections IV.A and 3.1.A.1 of Parts 1 and 3 of the licensee's Fire Hazard Analysis submittal, dated October 1, 1976 and April 1977. The NRC's position, as restated stated in Section 3.1.A.1 of this document establishes guidance on implementation of basic criteria for fire protection program organization and personnel staffing.

In response to the NRC's position discussed in Section A.1 of Appendix A to NRC Branch Technical Position APCSB 9.5-1 concerning the qualification requirements for the Fire Protection Engineer who will assist in various aspects of fire protection program development for the operating plant, the licensee states "comply" in Section 3.1.A.1 of the Fire Hazard Analysis submittal. The licensee further states, in part, "CECo has a Fire Protection Coordinator who reports to the Supervisor of Safety . . . Responsibilities of the Fire Protection Coordinator are: coordination of activities; procurement of equipment, resolve questions on standards and technical issues; make recommendations for improvements; coordinate, plan, and conduct inspections (make inspections of Dresden, Units 2 and 3, once a month); ensure that adequate fire fighting equipment is provided and that such equipment is maintained in good operating condition, coordinate with offsite fire department; conduct normal and preoperational testing; provide forms and instructions for reporting fires; issue publications outlining employee policy and procedures in fire protection; assist and supervise training of personnel; assist and advise departments concerned with established rules and standards; coordinate with the staff all matters of mutual concern and make final recommendations for specific actions to be taken on fire protection issues."

The inspectors identified the following examples of the licensee's failure to consistently and effectively comply with the staffing requirements for fire protection program implementation:

a. <u>Fire Protection Engineer</u>

A qualified Fire Protection Engineer was not involved in the development of certain aspects of the fire protection program for the operating plant as required by Section 3.1.A.1 of the licensee's Fire Hazard Analysis submittal. The qualifications for this individual were not stated in any document. The resume of the individual performing the original Fire Hazard Analysis is contained in Attachment 2A of the Fire Hazard Analysis submittal, but this individual is no longer employed by the licensee.

According to the licensee, there was a contract with M&M Protection Consultants which included services that would satisfy some of the responsibilities of the Fire Protection Engineer, but this contract expired in December 1984.

Although the licensee has employed another qualified Fire Protection Consultant firm to do some specific fire protection work relative to upgrading the fire protection program, this firm was not retained to fulfill all of the responsibilities of the Fire Protection Engineer.

b. <u>Fire Protection Coordinator</u>

The Fire Protection Coordinator was not performing all the duties at the site that are delineated in Section 3.1.A.1 of the licensee's Fire Hazard Analysis submittal. According to the licensee's staff, the individual that was originally assigned these duties was transferred to Corporate Quality Assurance some time ago. Once vacated, this position was not filled. The duties and responsibilities of the position were delegated to the Fire Marshal and other individuals within the CECo organization.

Through Amendment No. 86 to Facility Operating License No. DPR-19 (Unit 2) and Amendment No. 79 to Facility Operating License No. DPR-25 (Unit 3), the NRC accepted a proposed licensee staffing change. Figure 6.1-1 (Corporate and Station Organization Chart) shows a Fire Protection Inspector reporting to the Corporate Director of Quality Assurance Operations. The inspectors requested, but the licensee did not provide the inspectors with documentation to verify that the NRC was aware that the same individual who was the site Fire Protection Coordinator was filling the position entitled "Fire Protection Inspector" for Corporate Quality Assurance.

The licensee's failure to adhere to the staffing requirements discussed above resulted in programmatic breakdowns that have decreased the level of fire protection that was intended to satisfy Criterion 3 of Appendix A to 10 CFR 50. For example:

(1) A fire detection system was not installed on the refueling floor as required by Amendment No. 33 to Facility Operating License No. DPR-25. (This is discussed in Paragraph 7.a of the report.)

(2) Installed fire protection hardware and equipment was not being properly maintained. (This is discussed in Paragraphs 2.d, 2.e, 4, 5, and 7 of the report.)

(3) Technical specification surveillance procedures did not incorporate appropriate testing of quality affecting parameters in accordance with design and governing code requirements. (This is discussed in Paragraph 4 of the report.)

(4) Administrative controls did not adequately control fire protection features. (This is discussed in Paragraph 5 of the report.)

(5) Many deficiencies that were identified in LERs, NRC inspections, QA audits, and QA surveillances did not receive prompt or effective corrective action. (This is discussed in Paragraph 2 and 6 of the report.)

(6) Weaknesses in the scheduling of fire drills were identified. (This is discussed in Paragraphs 2.m, and 2.o of this report).

Failure to comply with the staffing requirements for development and implementation of the fire protection program is considered a violation of 10 CFR 50.48 and Criterion 3 of Appendix A to 10 CFR 50 (237/85033-01; 249/85029-01(DRS)).

The Station Fire Marshal's qualifications include 58 junior college credits in fire science; an associates degree in electronics engineering and 15 years experience as a volunteer firefighter. He has held the position of station fire marshal for seven years. At the present time, the fire marshal is assigned the following responsibilities:

a. Coordinate and assist in fire systems periodic testing.

b. Plan, coordinate, conduct, and critique fire drills.

c. Fire Brigade classroom training.

d. Review, revise, and write new administrative procedures.

e. Review, revise and write new surveillance procedures. Make work requests to repair deficiencies, verify surveillances are completed as required and maintain files on completed surveillances.

f. Review plant modifications, assist in training, testing, and development of procedures.

g. Maintain fire equipment, verify availability of spare parts and procurement of parts.

h. Participates in insurance inspections, Technical Specification Reviews, QA, INPO, and NRC audits.

i. Assure Technical Specification compliance.

j. Review work requests.

k. Verify fire watch and insurance notification.

l. Coordinate activities with the offsite fire department.

m. Make reports on deviations and fire damage experiences.

n. Perform plant cleanliness inspections.

o. Correspond with other agencies on fire protection issues.

p. Assure that the fire protection program meets NRC and other requirements.

q. Explain fire protection requirements to the licensee's staff when required.

According to the licensee's staff and Station Nuclear Engineering Department (SNED) procedure number PE Q.44, a qualified corporate fire protection engineer reviews new plant modifications prior to implementation by the Architect-Engineering firm. This appears to be the extent of the corporate fire protection engineer's involvement. The qualifications of the Station Fire Marshal do not appear to be commensurate with the list of responsibilities assigned to that position. This lengthy list of responsibilities constitute a workload that may not be achievable by a single individual, regardless of the individual's qualification and experience.

To resolve this concern, the licensee is requested to provide at the site, a written evaluation (complete work study) of the responsibilities assigned to the station fire marshal. This evaluation should make a determination of the fire marshal's ability to effectively achieve each delegated responsibility based on his qualifications and time constraints.

This is considered an Unresolved Item (237/85033-02; 249/85029-02(DRS)) pending Region III's review of this evaluation.

4. Technical Specification Surveillance Review

Technical Specification 6.2.A.11 requires that detailed written procedures be developed, approved and adhered to for implementation of the Fire Protection Program. The inspector's review of the licensee's surveillance procedures and test results for fire protection Technical Specification surveillance requirements resulted in identification of the following discrepancies:

a. Testing of Diesel Fire Pump at Least Once Per Operating Cycle

Section 4.12.B.1.(e) of Technical Specification No. 3.12.B requires that the station diesel fire pumps be demonstrated operable by performing a system functional test which includes simulated automatic actuation of the pumps throughout their operating sequence. The licensee's commitment in Section 3.5.E.2 of the Fire Hazard Analysis Report dated April, 1977, requires the fire pump installations to conform to NFPA standard No. 20. This commitment states that a plant modification would provide an adequate flow gage for full flow testing of the pumps in accordance with NFPA standard 20. The licensee's surveillance procedure Nos. DFPP 4124-3 and DFPP 4124-4 were deficient in that:

(1) The procedure required manual throttling of the pumps to achieve the specific flows contained in Technical Specification 3.12.B. and did not address automatic activation.

(2) The procedures required testing the pumps to the specific head and flow contained in the Technical Specification No. 3.12.B,

but failed to require testing for head and flow as specified in NFPA 20.

(3) Measurement of quality affecting parameters such as pump vibration under full flow conditions were not included in the test procedure or the test results.

(4) The test results were not compared to the original manufacturer's shop test curve or field acceptance test for the pumps because neither of these curves were available to the licensee's staff.

b. Testing of Water Suppression Systems at least Once Per Operating Cycle

Section 4.12.B.1.(e) of Technical Specification No. 3.12.B requires that fire suppression water systems be demonstrated operable by performing a system functional test which includes simulated automatic actuation of the systems throughout their operating sequence. The licensee's commitment in Section 3.5.E.3 of the Fire Hazard Analysis Report requires that automatic sprinkler systems conform to NFPA Standard No. 13.

The licensee's surveillance procedure No. SP 84-6-39 failed to incorporate appropriate test requirements to demonstrate the sprinkler system is operable in accordance with NFPA 13 in that:

(1) The procedure did not require flow from the inspector's test valve of wet sprinkler systems. Instead, the alarm bypass valve was used for this test.

(2) The procedure did not require flow from the two inch drain valve of wet or dry systems. Instead, the alarm bypass valve was used for this test.

c. Semiannual Testing of Fire Detectors

Section 4.12.A of Technical Specification No. 3.12.A requires that the fire detection system be demonstrated operable by performing a channel functional test every six months. The licensee commitment in Section 3.5.E.1 of the Fire Hazard Analysis Report requires that the fire detector system conform to the requirements of NFPA Standard 72D.

The licensee's surveillance procedure No. DFPP 4185-2 (Revision 4) failed to incorporate the following quality affecting parameters as required by NFPA 72D:

(1) Periodic cleaning of detector units.

(2) Periodic adjustment for sensitivity (Section 3.1.2 of the original SER required this test to be conducted).

According to the licensee's staff, an independent fire protection consultant has been employed to review all technical specification procedures and test results to evaluate their adequacy in accordance with NFPA standards and design requirements. This assessment was in progress at the time of the inspection and is expected to be completed by the end of 1985. According to the licensee, where necessary, the procedures will be revised to coincide with the governing code and design requirements.

This is considered an Open Item (237/85033-03; 249/85-029-03(DRS)) pending Region III's review of the licensee's actions.

No violations or deviations were identified.

5. Administrative Controls - Control of Welding, Cutting, and Ignition Sources

Licensee procedure No. DAP 3-11 (Revision 4) contained what appears to be acceptable instructions for controlling storage of flammable and combustible liquids, storage of compressed gas cylinders, and accumulation of rubbish and transient combustibles such as wood scaffolding, etc. The procedure specifies housekeeping and cleaning responsibilities to be followed by all employees and contractors.

No violations or deviations were identified in this area, however; the inspectors cautioned the licensee on a proposed revision to welding and cutting procedure No. DMP 4100-1 that would include a provision to facilitate ALARA concerns in high radiation areas. The inspectors informed the licensee that any relief from the requirements for a firewatch to remain in the immediate area thirty minutes after cutting and welding has been completed would have to be discussed with NRR.

6. Quality Assurance Program

The licensee's commitment to Quality Assurance for fire protection is documented in Section 3.3 of "Information Relevant to Fire Protection Systems and Programs" and in letters to the NRC on this subject dated January 24, February 24, March 20, and March 27, 1978, January 31 and April 30, 1979.

The inspectors review of the licensee's Quality Assurance Program for Fire Protection included review of the following:

a. Eleven criteria applicable to fire protection that satisfy Appendix A to Branch Technical Position 9.5-1 and supplement guidance "Nuclear Plant Functional Responsibilities, Administrative Controls and Quality Assurance."

b. Quality Assurance Surveillance Reports dated September 3-6, 1985, September 5-9, 1985, September 9-13, 1985, and September 16-30, 1985.

c. Annual Quality Assurance Audits Nos. QAA 12-84-I dated April 17, 1984, and QAA 12-83-I dated April 15, 1983.

d. Triennial Audit by M&M Protection Consultants dated December 4, 1984.

No violations or deviations were identified; however, the inspectors suggested to the licensee that for clarification, the statements made in Section 3.3 of the "Information Relevant to Fire Protection Systems and Programs" should be modified to indicate their specific commitment to a QA program to fire protection. As written, this statement can be interpreted to mean that the licensee committed to apply all of the criteria of Appendix B in 10 CFR 50 to fire protection.

The inspectors determined that the licensee's practice of considering fire protection as reliability-related is acceptable because this practice ensures that all of the eleven criteria contained in the NRC's Guidance are included in the program. In addition, this practice allows for the normal QA program for safety-related systems to be applied to fire protection in it's entirety. Only one QA manual exists for reliability-related systems and fire protection systems.

Although the licensee's Quality Assurance Program appears to be effectively identifying issues that are contributing to hardware and programmatic weaknesses, the licensee does not appear to be taking prompt and effective corrective actions. This is exemplified by the remaining open items that have been identified in QA audits and surveillances, LERs, and NRC inspections. (This is further discussed in 3.b.(5) of the report.)

7. Plant Tours

During tours of the plant, the inspectors observed the following deficient conditions:

a. Failure to Comply with License Condition No. 2.B. of Amendment No. 33 to Facility Operating License No. DPR-25 and Amendment No. 36 to Provisional Operating License No. DPR-19.

Section 5.1.6.6 of the original Fire Protection SER for Dresden Units 2/3 dated March 22, 1978 states that the licensee proposed the installation of an automatic fire detection system to provide early warning of a fire in the Refueling Floor Area in order to satisfy the objectives of Criterion 3 of Appendix A to 10 CRF 50. Amendment No. 36 to Provisional Operating License No. DPR-19 (Unit 2) and Amendment No. 33 to Facility Operating License No. DPR-25 (Unit 3) dated October 1, 1980, require that the early warning automatic fire detection system for the refueling floor area be installed by start up following the 1979 Unit 3 refueling outage.

As of the date of this inspection (approximately six years after start up following the Unit 3 1979 refueling outage) the licensee has failed to comply with the provisions of Amendment No. 36 to Provisional Operating License No. DPR-19 and Amendment No. 33 to Facility Operating License No. DPR-25. An early warning automatic fire detection system fire detection system has not been installed

in the Refueling Floor Area and no compensatory measures have been taken as a result of this decreased effectiveness of the plant's fire protection features.

The installation of an automatic early warning fire detection system in the refueling floor area was not discussed in any of the licensee's correspondence to the NRC that requested amendments to modify the plant's fire protection Technical Specifications to incorporate Limiting Conditions for Operation and Surveillance Requirements for the fire protection modifications required by the original SER for Dresden Units 2/3. None of the proposed Tables 3.12.1 to Technical Specification 3.12 listed fire detection instruments in the refueling floor area. However, sufficient information existed which should have alerted the licensee that he was in violation of a license condition. For example:

(1) By letter dated February 25, 1980 (R. F. Janecek-CECO to T. A. Ippolito-NRC) the licensee noted that they did not believe installation of an automatic early warning fire detection system in the refueling floor area was warranted based on low fire loading and the ability to make up water and cool the spent fuel pools in the event of a loss of either Unit's spent Fuel pool cooling equipment due to fire. This letter did not request relief from the installation of a refueling floor fire detection system. No official NRC response was issued for this letter.

(2) By letter dated March 18, 1980 (L. Derderian-NRC to M. Antonetti - Gage Babcock and Associates - Consultants to the Licensee) the NRC referenced a March 17, 1980 telecon record with T. Pickens (CECO) in which the following was agreed to concerning Reactor Building Refueling floor fire detection systems for Dresden Units 2/3 and Quads Cities Units 1 and 2:

   (a) The license was to confirm to the NRC that in the most heavy fire loading situations (i.e. refueling periods), the loading would not exceed that necessary to cause structural failures.

   (b) The licensee was to confirm that structural concrete protection extends from the floor to some specified height, lessening the likelihood of structural failure.

   (c) The licensee was to recalculate average combustible loading subtracting out the pool areas.

   The licensee could not provide the inspectors with documented evidence that these issues were addressed.

This failure to followup on implementation of a license condition is indicative of a programmatic breakdown which has resulted in a reduced level of fire protection than was intended to satisfy

criterion 3 of Appendix A to 10 CFR 50 and is considered a violation of Amendment No. 36 to Provisional Operating License No. DPR-19, Amendment No. 33 to Facility Operating License No. DPR-25, 10 CFR 50 (237/85-033-04; 249/85-029-04)(DRS).

b. Preparations for the Upcoming Extended Unit 3 Outage Separation of Unit 1 from Units 2/3

During plant tours and in meetings with the licensee during the inspection, the licensee agreed to update their response to the NRC and describe the administrative controls and the actions that will be necessary to isolate Unit 1 from Units 2 and 3 since Unit 1 is no longer operational but shares common areas with Units 2 and 3. The inspectors also requested that the licensee describe those administrative controls and actions that will be necessary to separate common areas in Units 2/3 while Unit 2 is operating and Unit 3 is in an extended outage.

This is considered an Open Item (237/85-033-05; 249/85-029-05)(DRS) pending further review by Region III.

c. Self Contained Breathing Air Supply for the Fire Brigade

Section 3.4.D.4(h) of the document entitled "Information Relevant to Fire Protection Systems and Programs", requires that breathing apparatus using full face piece positive pressure masks that are approved by NIOSH be provided for the fire brigade.

The inspectors examined the fire brigade Scott Air Pak breathing air cylinders that were provided on Fire Chart No. 2. Four out of four of these cylinders contained 1800 pounds of air pressure. According to the licensee's staff, a minimum of 2200 pounds of air pressure should be contained in each cylinder. 2400 pounds of air pressure would indicate the cylinder is full and may provide a 30 minute air supply for the average fire brigade member. The cylinder gauges have a range of up to 3000 pounds of air pressure.

A December 1984 three year audit recommended that a set of written instructions be provided at the breathing air cylinder filling station to assure that the cylinders are filled properly. Filling of the cylinders is the responsibility of Health Physics. Due to time constraints, the inspectors were unable to contact Health Physics to follow up this concern. Therefore, the licensee is requested to provide at the site the appropriate acceptance criteria for filling breathing air supply cylinders. This is considered an Open Item (237/85-033-06; 249/85-049-06)(DRS) pending Region III review of the licensee's breathing air cylinder filling procedures.

d. 300 Pound Fixed Cardox System Supply Tank First Floor, Turbine Building

During plant tours, the inspectors observed the following deficiencies on the main CO_2 system storage tank located on the first floor of the turbine buildings.

(1) The access door to the tank compressor motor was missing.

(2) The glass cover to the tank's mercoid switch located inside the access door was missing.

The licensee had no explanation for these deficiencies, but agreed to take immediate corrective actions.

This is considered an Open Item (237/85-033-07; 249/83-029-07)(DRS) pending further verification of the licensee's corrective actions by Region III.

8. Open Items

Open items are matters which have been discussed with the licensee, which will be reviewed further by the inspector, and which involve some action on the part of the NRC of licensee of both. Open items disclosed during the inspection are discussed in Paragraphs 4.c, 5.a, 7.b, 7.c, 7.d.

9. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, items of noncompliance, or deviations. An unresolved item disclosed during the inspection is discussed in Paragraph 3.c.

10. Exit Interview

The inspectors met with the licensee representatives at the conclusion of the inspection on October 4, 1985, and summarized the scope and findings of the inspection. The licensee acknowledged the statements made by the inspectors. The inspectors also discussed the likely informational content of the inspection report with regard to documents reviewed by the inspector during the inspection. The licensee did not identify any such documents as proprietary. On October 21, 1985, in a telephone conversation with the licensee additional concerns regarding the lack of fire detectors on the refueling floor were discussed with the licensee.

85033 / 85029

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

DEC 26 1985

ORIG: FILE/IE INSP
CC: D SCOTT
CC: D ADAM

Docket No. 50-237
Docket No. 50-249

854943

Commonwealth Edison Company
ATTN: Mr. Cordell Reed
Vice President
Post Office Box 767
Chicago, IL 60690

REC'D DEC 27 1985

Gentlemen:

By letter dated November 14, 1985, (copy enclosed) Region III transmitted to you Inspection Reports 50-237/85033(DRS) and 50-249/85029(DRS). These reports documented the results of an unannounced safety inspection conducted between September 30 and October 21, 1985, to establish the adequacy of fire protection program implementation at Dresden Nuclear Power Station, Units 2 and 3. As discussed in the November 14 letter, two issues were identified during the inspection which were under consideration for escalated enforcement actions. These issues were failure to install fire detectors on the refueling floor as required by license condition and failure to effectively implement the fire protection program in accordance with 10 CFR 50.48 as evidenced by numerous and, in some cases, recurring deficiencies.

On November 19, 1985, an enforcement conference was held in the Region III office with you and members of your staff to review these issues and obtain additional information regarding their significance. A list of attendees is contained in Enclosure 2 to this letter. During this conference, Region III management expressed concerns relative to your failure to satisfy an explicit license condition requirement and your apparent failure to provide sufficient resources to effectively implement the Dresden Fire Protection Program.

In response to the concern expressed over your failure to comply with an explicit license condition, you presented information demonstrating that (1) failure to install fire detectors on the refueling floor was of minor technical significance based on the low fire loading and the lack of safe shutdown equipment in that area; (2) the Office of Nuclear Reactor Regulation had been informed in a letter dated February 25, 1980, that fire detectors were not necessary on the refueling floor; and (3) you had undertaken a review of regulatory, commitment, and code compliance at your operating stations which had identified other issues requiring resolution and would likely have identified the failure to install the subject fire detectors. It was your contention that item (2) above demonstrated that you were aware of and sensitive to the license condition requiring the installation of refueling floor fire detectors and that item (3) demonstrated your commitment to ensuring that all required fire protection features had been implemented.

In response to the concern expressed over your apparent failure to effectively implement the fire protection program at Dresden, you presented information on the existing fire protection staffing and experience at Dresden but indicated that the matter was under review and that additional fire protection expertise may be indicated.

Your response to the issues discussed in the November 19 Enforcement Conference was supplemented in a letter dated December 2, 1985, submitted to the Office of Nuclear Reactor Regulation discussing the preliminary results of your review of the status of compliance with fire protection requirements at Dresden and Quad Cities. This letter identified several outstanding deficiencies, proposed methods for resolution, established completion dates, and requested approval of the proposed resolutions.

Region III reviewed the information presented at the November 19 Enforcement Conference and contained in your December 2 letter and has reached the following conclusions:

1. Failure to install fire detectors on the refueling floor and failure to effectively implement your fire protection program at Dresden are violations of NRC regulations.

2. Failure to satisfy a license condition is of significant regulatory concern; however, you have demonstrated that, in the case of the refueling floor fire detectors, the safety significance is low and that you were actively pursuing a program to ensure that compliance would have been achieved. Additionally, your December 2, 1985, letter provides us assurances that this and similar issues are being aggressively pursued in a timely fashion.

3. With regard to the failure to effectively implement the fire protection program at Dresden, you demonstrated that you had previously identified concerns in that area and were pursuing resolution of those concerns. During the enforcement conference, you verbally committed to bring additional resources to bear in this area.

Based on the above, it is concluded that while escalated enforcement action is not warranted for your failure to install fire detectors on the refueling floor and your failure to effectively implement the Dresden Fire Protection Program, issuance of a Notice of Violation is appropriate. Accordingly, Enclosure 1 to this letter transmits to you a Notice of Violation for which a written response is required.

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

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

Sincerely,

James G. Keppler

James G. Keppler
Regional Administrator

Enclosures:
1. Notice of Violation
2. November 19, 1985 Enforcement Conference Attendance List
3. Letter dtd 11/14/85, NRC to Commonwealth Edison Co.

cc w/enclosures:
D. L. Farrar, Director of Nuclear Licensing
D. J. Scott, Plant Manager
DCS/RSB (RIDS)
Licensing Fee Management Branch
Resident Inspector, RIII
Phyllis Dunton, Attorney General's Office, Environmental Control Division

ENCLOSURE 1

NOTICE OF VIOLATION

Commonwealth Edison Company

Docket No. 50-237
Docket No. 50-249

As a result of the inspection conducted on September 30 through October 21, 1985, and in accordance with the "General Policy and Procedures for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (1985), the following violations were identified:

1. Amendment No 36 to Provisional Operating License No. DPR-19 and Amendment No. 33 to Facility Operating License No. DPR-25 require the licensee to complete the modifications identified in Paragraphs 3.1.1 through 3.1.23 of the NRC's Fire Protection Safety Evaluation dated March 1978 by startup following the 1979 Unit 3 refueling outage. Paragraph 3.1.1 subparagraph (6) of the NRC's Fire Protection Safety Evaluation dated March 1978 states that early warning Fire Detection Systems will be provided for the Reactor Building refueling floor.

   Contrary to the above, during the period September 30 through October 21, 1985, it was identified that an early warning fire detection system was not installed on the Reactor Building refueling floor. Further, it was determined that an early warning fire detection system had never been installed on the refueling floor.

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

2. 10 CFR 50.48(a) requires that each operating nuclear power plant have a fire protection plan that satisfies Criterion 3 of Appendix A to 10 CFR Part 50. It further requires that the plan shall describe specific features necessary to implement the program such as administrative controls and personnel requirements to limit fire damage to structures, systems, or components important to safety so that the capability to safely shut down the plant is ensured.

   Section 3.1.A.1 of the licensee's Fire Hazards Analysis Submittal, which forms part of the licensee's approved fire protection program, states that the licensee has a Fire Protection Coordinator whose responsibilities include, in part, program coordination, equipment procurement, program enhancement, conducting inspections, and supervising training of personnel.

   Contrary to the above, the licensee has failed to consistently and effectively staff the Fire Protection Coordinator position with the result that certain fire protectin equipment was not installed, hardware and equipment were not being properly maintained, required training was not completed, and prompt and effective corrective action was not taken for identified deficiencies.

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

With respect to Item 1, information provided after the inspection showed that action had been taken to resolve the identified violation and to prevent recurrence. Consequently, no reply to this violation is required and we have no further questions regarding this matter. With respect to Item 2, pursuant to the provisions of 10 CFR 2.201, you are required to submit to this office within thirty days of the date of this Notice a written statement or explanation in reply for the violation: (1) corrective action taken and the results achieved; (2) corrective action to be taken to avoid further violations; and (3) the date when full compliance will be achieved. Consideration may be given to extending your response time for good cause shown.

December 26, 1985
Dated

James G Keppler
James G. Keppler
Regional Administrator

ENCLOSURE 2

NOVEMBER 19, 1985 ENFORCEMENT CONFERENCE ATTENDANCE LIST

A. Commonwealth Edison Company Personnel

B. B. Stephenson, Division Vice President
C. Reed, Vice President of Nuclear Operations
L. Del George, Assistant Vice President, Engineering and Licensing
J. L. Reed, Quad Cities Fire Protection Coordinator
G. Spedl, Assistant Superintendent of Technical Services, Quad Cities
J. J. McDonald, Station Nuclear Engineering Department Fire Protection Coordinator
J. Achterberg, Technical Staff Supervisor, Dresden
J. Bitel, Operations Quality Assurance Manager
P. F. Hart, Quality Assurance Fire Protection Engineer
P. A. Lau, Quality Assurance Supervisor, Dresden
J. Wojnarowski, Nuclear Licensing Administrator
L. Davis, Supervisor of Station Support Services
T. G. Hausheer, Support Services Fire Protection Engiener
L. F. Gerner, Regulatory Assurance Superintendent
D. J Scott, Station Manager, Dresden
J. D. Brunner, Assistant Superintendent of Technical Services, Dresden
M. Turnback, Operating Plant Licensing Director
R. Rybak, Station Nuclear Engineering Department Fire Protection Supervisor
J. W. Dingler, Senior Licensing Project Engineer - Sargent and Lundy

B. U. S. Nuclear Regulatory Commission Personnel

A. B. Davis, Deputy Regional Administrator, Region III
C. J. Paperiello, Director, Division of Reactor Safety, Region III
B. A. Berson, Regional Counsel, Region III
L. A. Reyes, Chief, Operations Branch, Region III
W. G. Guldemond, Chief, Operational Programs Section, Region III
C. B. Ramsey, Reactor Inspector, Region III
A. L. Madison, Senior Resident Inspector, Quad Cities, Region III
L. G. McGregor, Senior Resident Inspector, Dresden, Region III
W. H. Schultz, Enforcement Coordinator, Region III
R. A. Gilbert, Dresden Project Manager, Office of Nuclear Reactor Regulation

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

NOV 14 1985

Docket No. 50-237 85-33
Docket No. 50-249 85-29

Commonwealth Edison Company
ATTN: Mr. Cordell Reed
Vice President
Post Office Box 767
Chicago, IL 60690

Gentlemen:

This refers to the routine safety inspection conducted by Messrs. J. Holmes and C. Ramsey of this office on September 30 through October 21, 1985, of activities at Dresden Nuclear Power Station, Units 2 and 3, authorized by NRC Operating Licenses No. DPR-19 and No. DPR-25 and to the discussion of our findings with Mr. D. Scott at the conclusion of the inspection.

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.

During this inspection, certain of your activities appeared to be in violation with NRC requirements. These issues, identified in paragraphs 3 and 7.a of the enclosed inspection report, are being reviewed for potential escalated enforcement action. The results of that review will be forwarded to you by separate correspondence which will identify the nature of expected formal response.

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

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

NOV 14 1985

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

Sincerely,

Carl J Paperiello

Carl J. Paperiello, Director
Division of Reactor Safety

Enclosure: Inspection Reports
No. 50-237/85033(DRS); and
No. 50-249/85029(DRS)

cc w/enclosure:
D. L. Farrar, Director
of Nuclear Licensing
D. J. Scott, Plant Manager
DCS/RSB (RIDS)
Licensing Fee Management Branch
Resident Inspector, RIII
Phyllis Dunton, Attorney
General's Office, Environmental
Control Division

RIII Ramsey 11/14/85 | RIII Holmes 11/14/85 | RIII Guldemond 11/14/85 | RIII Wright 11/14/85 | RIII Reyes 11/14/85 | RIII Paperiello 11/14/85

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-237/85033(DRS); 50-249/85029(DRS)

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

Licensee: Commonwealth Edison Company
P. O. Box 767
Chicago, IL 60690

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

Inspection At: Morris, IL

Inspection Conducted: September 30 through October 21, 1985

Inspectors: J. Holmes — 11-13-85 Date

C. Ramsey — 11-13-85 Date

Approved By: W. G. Guldemond, Chief
Operational Programs Section — 11-13-85 Date

Inspection Summary

Inspection on September 30 through October 21, 1985 (Reports No. 50-237/85033(DRS); 50-249/85029(DRS))

Areas Inspected: Routine, unannounced safety inspection conducted to verify the adequacy of the facility's fire protection program implementation and to determine the status of LERs and previous open items. The inspection involved 71 inspector-hours by two NRC inspectors including 2 inspector-hours onsite during off-shifts and 11 inspector-hours conducting in-office review at the Region III office.

Results: Of the 6 areas inspected, no violations or deviations were identified in four areas. Two violations were identified in the remaining two areas (failure to adhere to program staffing requirements - Paragraph 3; failure to comply with a license condition to install an automatic fire detection system in the Reactor Building refueling floor area - Paragraph 7a).

DETAILS

1. Persons Contacted

DNPS

D. Adam, Compliance Administrator
*J. Brunner, Assistant Superintendent, Technical Services
T. Ciesla, Assistant Superintendent, Operations
*M. Dillon, Fire Marshall
*R. Flissner, Service Superintendent
*T. Hausheer, Nuclear Services, Technical
*P. Lau, QA Supervisor
*J. McDonald, Station Nuclear Engineering
*B. Rybak, Station Nuclear Engineering
*D. Scott, Station Manager
*R. Whalen, Technical Staff
J. Wujciga, Production Superintendent

US NRC

E. Hare, Resident Inspector
*L. McGregor, Senior Resident Inspector
S. Stasek, Resident Inspector

*Denotes those in attendance at the exit meeting of October 4, 1985.

2. Licensee Actions on Previous Inspection Findings

a. (Open) LER (237/85029) and Violation (237/85028-01): Auxiliary electric equipment room halon system declared inoperable due to ventilation dampers failing to close. Fire watch was not established per Technical Specification No. 3.12.H.2.

Region III's followup of this event is documented in Inspection Report No. 50-237/85028(DRS). As a result of this followup violation No. 237/85028-01 was issued. No response to this violation was required because the licensee's interim and long term corrective actions were determined satisfactory. The interim corrective actions were implemented prior to or during the followup inspection. The proposed long term corrective actions have not been implemented. Therefore, this event report remains open.

b. (Closed) LER (249/85014): Wet pipe sprinkler system in Unit 3 turbine trackway had to be rerouted to allow for overhead clearance for new turbine rotors. The sprinkler system was out of service 5 1/2 hours beyond the 14 day limit permitted by Technical Specification 3.12.C.3.

The event report is closed based on the licensee's corrective actions taken which included restoration of the system to service and functional testing prior to declaring the system operable.

c. (Closed) LER (237/85010): Fire door for the Unit 2 125V DC battery room found open. A fire watch was not established within one hour per Technical Specification 3.12.F.2.

This event report is closed based on the licensee's corrective actions taken which included immediate closure of the fire door and training/instruction of plant operators on the requirements to keep fire doors closed at all times when not in use.

d. (Open) LER (237/84-20): Two of seven root valves that were installed on fire hose stations were found to be in the closed position rendering the fire hose stations inoperable.

Although the licensee's corrective actions for this event included prompt opening of the closed root valves, the inspectors determined that the licensee's program for administratively controlling valves that are not electrically supervised using wire seals to secure these valves in the open position and performing monthly inspections to verify valve positions does not appear to be working. During the inspection the inspectors observed several non-electrically supervised valves in the fire protection system with missing or damaged wire seals.

To correct this problem, the licensee stated that the program for administratively controlling these valves is being upgraded to include locking these valves (chain and lock) in the open position in addition to monthly inspections to verify each valve position in accordance with NFPA Standard 26. This event report remains open pending Region III verification of the licensee's upgraded corrective actions.

e. (Open) LER (237/84-17; 237/84-05): Failure to establish continuous or hourly fire watch patrols due to inoperability of all or portions of fire detection and sprinkler alarm systems in the control room.

During these events and at the present time, the fire detection and sprinkler alarm system printer indications are interlocked (dependent) into the plant security system computer. Indication of fire detection and sprinkler alarms in the control room was lost in two events either because of a loss of power to the plant security system computer or because of modifications being made to upgrade the plant security system computer. Apparently, any failure of the plant security system computer can cause the loss of all or portions of fire detection and sprinkler alarm annunciation in the control room.

This installation does not comply with the licensee's commitment to NFPA 72D as stated in the licensee's April 1977 response (point-by-point comparison) to Appendix A to NRC Branch Technical Position

(BTP) APCSB 9.5-1. The licensee's scheduled plant modification No. M12-2/3-84-109 identifies corrective action for this problem as the installation of independent circuits for fire detection and sprinkler system alarms which alarm and annunciate in the control room in accordance with NFPA 72D. This modification is scheduled to be completed in December 1985. These LERs will remain open pending Region III verification of the licensee's corrective actions.

f. (Closed) LER (237/84-11): Fire wall penetrations to Unit 2/3 diesel generator rooms were not sealed.

This event report is closed based on the licensee's corrective actions taken, which include establishment of a fire watch within one hour per Technical Specification No. 3.12.F.2 and sealing the penetrations per drawing No. 12E-6058.

g. (Closed) LER (237/84-08): NRC inspection of the licensee's compliance with fire protection Technical Specification surveillance requirements identified that a cardox system master valve was not being tested in the automatic mode. The master valve test procedure was written to test the valve in the manual mode.

This event report is closed based on the licensee's corrective actions taken which included prompt removal of the master valve from service, revision of the surveillance test procedure, and satisfactory testing of the valve in the automatic mode.

h. (Closed) LER (249/83-34/03L): Unit 3 trackway sprinkler system out of service due to damage by mobile crane boom.

This event is closed based on the licensee's corrective actions taken, which included making the necessary system repairs, prompt restoration of the system to service and instructions to plant personnel regarding the movement of mobile cranes and the fragility of systems and components in their path.

i. (Closed) LER (249/83-17/03L): HPCI deluge system solenoid valve taken out of service because the valve would not reset.

This event report is closed based on the licensee's corrective actions taken, which included prompt removal of the deluge system from service, making the necessary repairs to the solenoid valve and restoration of the system to service.

j. (Closed) LER (237/81-15/03L): Unit 2/3 diesel generator room CO_2 system heat detector surveillances not performed per Technical Specification 4.12.A.1.

This event report is closed based on the licensee's corrective actions taken, which included instruction to plant personnel to perform the required heat detector surveillances and satisfactory performance of the surveillance.

k. (Closed) Violations (237/81-09-01; 249/81-06-01): Four penetration seals identified as being defective were inoperable for an excessive period of time. Neither prompt nor timely corrective action was taken.

This item is closed based on the licensee's June 29, 1981 response to Region III which discussed the licensee's corrective actions taken to avoid future violations in this area. The inspector's review of Procedure No. DFPP-4175-2, Revision 4, indicated that appropriate instructions are provided to plant personnel which refer to detail drawings for proper installation of penetration fire seals.

l. (Open) Violations (237/81-09-03; 249/81-06-03): (a) Fifty percent of fire extinguishers sampled did not have 1981 monthly inspection tags attached; (b) 5 year hydrostatic test for portable CO_2 extinguisher cylinders were overdue; and (c) numerous compressed gas cylinders were improperly stored.

The licensee's corrective actions identified in their June 29, 1981 response to items (a) and (b) of this violation were ineffective. Subsequent QA audits and surveillance by the licensee's onsite QA department have revealed that these deficiencies are continuing. For example, deficiencies identified in QA surveillance No. QAS 12-85-236 for the period September 23 through 29, 1985 include the following: wrong date on extinguisher tags; extinguishers past due for 5 year hydro testing; no service date on extinguisher tags; no seal on extinguisher pull pin; partially discharged extinguisher.

During plant tours by the inspectors, identical deficiencies were observed. In one instance, a CO_2 portable extinguisher hose was damaged to the extent that the webbing in the hose was exposed. A hole existed in the webbing that may have allowed the extinguishing agent to escape through the hose prior to reaching the CO_2 discharge nozzle. This extinguisher was located on fire cart No. 2. In addition, the inspectors observed that wheeled dry chemical extinguishing units Nos. PK 21 and PK 22 had tags which indicated that surveillances were missed the months of May and September 1985. The continuing existence of this type of deficiency is indicative of a lack of management attention in this area. Management attention and staffing is the subject of a violation documented in paragraph 3 of this report. Your response to that violation should address your corrective actions for failing to properly maintain fire extinguishers. These items will remain open pending the further review of the licensee's corrective actions by Region III.

Item (c) of this violation is closed based on the licensee's corrective action taken which included the installation of metal storage racks for compressed gas cylinders, securing the cylinders with metal chains and revision of Procedure No. DAP 3-11.

m. (Open) Unresolved Item (237/81-09-06; 249/81-06-06): Fire brigade drills and training do not appear to meet the intent of NRC requirements.

Section 6.0 of the original fire protection SER, dated March 1978, recommended that the licensee's administrative controls follow the guidelines set forth in the NRC Guidance Document entitled "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance." A supplement to the original SER was issued December 2, 1980. Section 3.1 of this supplemental SER closes out the issue of administrative controls with the NRC staff's acceptance of the licensee's discussion of administrative controls provided in letters dated January 24, February 24, March 20 and July 27, 1978, January 31, and April 30, 1979. Therefore, Section III.1.3.b of Appendix R is not applicable to administrative controls for fire protection at Dresden.

Based on the licensee's submittals discussed above, the NRC staff concluded that the licensee's administrative controls for fire protection met NRC guidelines and, the applicable regulatory requirement for fire protection administrative controls at Dresden is the Commission's guidance issued on the implementation of General Design Criterion 3 of Appendix A to 10 CFR Part 50 for existing power plants.

Section 2.0 and 3.0 of Attachment No. 2 to NRC Guidance Document "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance" requires practice sessions be held for fire brigade members to provide each brigade member with experience in actual fire extinguishment and the use of emergency breathing apparatus under strenuous conditions. Fire brigade drills are required to be performed so that the fire brigade can practice as a team. The drills are to be performed at regular intervals but not to exceed three months for each fire brigade. The drills are required to be critiqued to assess each brigade member's knowledge of his role in fire fighting strategy.

The licensee is not meeting these requirements for the following reasons:

(1) By attempting to meet the requirements contained in Section III.1.3.b of Appendix R to 10 CFR 50, the licensee has been conducting one fire drill per month with the intent of getting all designated fire brigade members involved in at least two drills per year.

(2) Practice sessions that provide each brigade member with actual fire extinguishment experience and the use of emergency breathing apparatus under strenuous conditions (full fire fighting gear) have not been conducted due to a breakdown in contractual arrangements with an independent firm.

(3) Fire brigade drills have not been critiqued at three year intervals by qualified individuals independent of the licensee's staff.

To resolve this concern, the licensee is requested to make available a detailed assessment of fire brigade drills, practice sessions and three year audits of fire brigade drills by qualified individuals independent of the licensee's staff. This assessment should establish whether the licensee is in compliance with commitments made to the NRC which resulted in the NRC staff conclusions that the licensee's administrative controls for fire protection were acceptable.

This item remains open pending region review of the licensee's assessment.

n. (Open) Unresolved Item (237/81-09-07; 249/81-06-07): Specific pre-fire fighting plans or strategies for all safety-related areas and areas presenting a hazard to safety-related equipment were not developed and implemented.

As discussed in item 237/81-09-06; 249/81-06-06 above, the requirements of 10 CFR 50, Appendix R (Sections III.K.11 and 12) are not applicable in this case. The applicable requirements are contained in Attachment No. 5 of NRC Guidance Document "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance."

During the inspection, the licensee provided the inspectors with a copy of pre-fire plans that contained specific fire fighting strategies for fighting fires in all safety-related areas and areas that present a hazard to safety-related equipment. The pre-fire plans appear to provide adequate fire fighting procedures and instructions. However, these plans have not been implemented. According to the licensee, the plans will be implemented and incorporated into fire brigade training lesson plans by the end of the first quarter of 1986. This item will remain open pending said implementation.

o. (Open) Unresolved Item (237/81-09-08; 249/81-06-08): Fire brigade practice sessions have not been conducted in accordance with commitments made to the NRC. A hands-on practice session was held in 1979 with full brigade attendance, but no practice session was held in 1980.

This item will remain open pending Region III review of the licensee's response to Item b of Unresolved Item No. 237/81-09-06; 249/81-06-06 as discussed in this report.

3. <u>Fire Protection Program Organization and Personnel Staffing</u>

10 CFR 50.48 requires that each operating nuclear power plant have a fire protection plan that satisfies Criterion 3 of Appendix A to 10 CFR 50. Except for the requirements of Section III.G, III.J, and III.O of Appendix R to 10 CFR 50, the approved fire protection plan that satisfies Criterion 3 of Appendix A to 10 CFR 50 is discussed in the original fire protection SER, dated March 1978, a fire protection SER Supplement, dated December 2, 1980, and the licensee's Fire Hazard Analysis submittals entitled

"Information Relevant to Fire Protection Systems and Programs" dated October 1976, January 1977, and April 1977. Furthermore, the licensee committed to follow certain NRC Supplemental Guidance Documents as discussed in letters to the NRC, dated January 24, February 24, March 20 and July 27, 1978; January 31 and April 30, 1979.

The requirements for overall responsibility for the Fire Protection Program are discussed in Sections IV.A and 3.1.A.1 of Parts 1 and 3 of the licensee's Fire Hazard Analysis submittal, dated October 1, 1976 and April 1977. The NRC's position, as restated stated in Section 3.1.A.1 of this document establishes guidance on implementation of basic criteria for fire protection program organization and personnel staffing.

In response to the NRC's position discussed in Section A.1 of Appendix A to NRC Branch Technical Position APCSB 9.5-1 concerning the qualification requirements for the Fire Protection Engineer who will assist in various aspects of fire protection program development for the operating plant, the licensee states "comply" in Section 3.1.A.1 of the Fire Hazard Analysis submittal. The licensee further states, in part, "CECo has a Fire Protection Coordinator who reports to the Supervisor of Safety . . . Responsibilities of the Fire Protection Coordinator are: coordination of activities; procurement of equipment, resolve questions on standards and technical issues; make recommendations for improvements; coordinate, plan, and conduct inspections (make inspections of Dresden, Units 2 and 3, once a month); ensure that adequate fire fighting equipment is provided and that such equipment is maintained in good operating condition, coordinate with offsite fire department; conduct normal and preoperational testing; provide forms and instructions for reporting fires; issue publications outlining employee policy and procedures in fire protection; assist and supervise training of personnel; assist and advise departments concerned with established rules and standards; coordinate with the staff all matters of mutual concern and make final recommendations for specific actions to be taken on fire protection issues."

The inspectors identified the following examples of the licensee's failure to consistently and effectively comply with the staffing requirements for fire protection program implementation:

a. <u>Fire Protection Engineer</u>

A qualified Fire Protection Engineer was not involved in the development of certain aspects of the fire protection program for the operating plant as required by Section 3.1.A.1 of the licensee's Fire Hazard Analysis submittal. The qualifications for this individual were not stated in any document. The resume of the individual performing the original Fire Hazard Analysis is contained in Attachment 2A of the Fire Hazard Analysis submittal, but this individual is no longer employed by the licensee.

According to the licensee, there was a contract with M&M Protection Consultants which included services that would satisfy some of the responsibilities of the Fire Protection Engineer, but this contract expired in December 1984.

Although the licensee has employed another qualified Fire Protection Consultant firm to do some specific fire protection work relative to upgrading the fire protection program, this firm was not retained to fulfill all of the responsibilities of the Fire Protection Engineer.

b. <u>Fire Protection Coordinator</u>

The Fire Protection Coordinator was not performing all the duties at the site that are delineated in Section 3.1.A.1 of the licensee's Fire Hazard Analysis submittal. According to the licensee's staff, the individual that was originally assigned these duties was transferred to Corporate Quality Assurance some time ago. Once vacated, this position was not filled. The duties and responsibilities of the position were delegated to the Fire Marshal and other individuals within the CECo organization.

Through Amendment No. 86 to Facility Operating License No. DPR-19 (Unit 2) and Amendment No. 79 to Facility Operating License No. DPR-25 (Unit 3), the NRC accepted a proposed licensee staffing change. Figure 6.1-1 (Corporate and Station Organization Chart) shows a Fire Protection Inspector reporting to the Corporate Director of Quality Assurance Operations. The inspectors requested, but the licensee did not provide the inspectors with documentation to verify that the NRC was aware that the same individual who was the site Fire Protection Coordinator was filling the position entitled "Fire Protection Inspector" for Corporate Quality Assurance.

The licensee's failure to adhere to the staffing requirements discussed above resulted in programmatic breakdowns that have decreased the level of fire protection that was intended to satisfy Criterion 3 of Appendix A to 10 CFR 50. For example:

(1) A fire detection system was not installed on the refueling floor as required by Amendment No. 33 to Facility Operating License No. DPR-25. (This is discussed in Paragraph 7.a of the report.)

(2) Installed fire protection hardware and equipment was not being properly maintained. (This is discussed in Paragraphs 2.d, 2.e, 4, 5, and 7 of the report.)

(3) Technical specification surveillance procedures did not incorporate appropriate testing of quality affecting parameters in accordance with design and governing code requirements. (This is discussed in Paragraph 4 of the report.)

(4) Administrative controls did not adequately control fire protection features. (This is discussed in Paragraph 5 of the report.)

(5) Many deficiencies that were identified in LERs, NRC inspections, QA audits, and QA surveillances did not receive prompt or effective corrective action. (This is discussed in Paragraph 2 and 6 of the report.)

(6) Weaknesses in the scheduling of fire drills were identified. (This is discussed in Paragraphs 2.m, and 2.o of this report).

Failure to comply with the staffing requirements for development and implementation of the fire protection program is considered a violation of 10 CFR 50.48 and Criterion 3 of Appendix A to 10 CFR 50 (237/85033-01; 249/85029-01(DRS)).

The Station Fire Marshal's qualifications include 58 junior college credits in fire science; an associates degree in electronics engineering and 15 years experience as a volunteer firefighter. He has held the position of station fire marshal for seven years. At the present time, the fire marshal is assigned the following responsibilities:

a. Coordinate and assist in fire systems periodic testing.

b. Plan, coordinate, conduct, and critique fire drills.

c. Fire Brigade classroom training.

d. Review, revise, and write new administrative procedures.

e. Review, revise and write new surveillance procedures. Make work requests to repair deficiencies, verify surveillances are completed as required and maintain files on completed surveillances.

f. Review plant modifications, assist in training, testing, and development of procedures.

g. Maintain fire equipment, verify availability of spare parts and procurement of parts.

h. Participates in insurance inspections, Technical Specification Reviews, QA, INPO, and NRC audits.

i. Assure Technical Specification compliance.

j. Review work requests.

k. Verify fire watch and insurance notification.

l. Coordinate activities with the offsite fire department.

m. Make reports on deviations and fire damage experiences.

n. Perform plant cleanliness inspections.

o. Correspond with other agencies on fire protection issues.

p. Assure that the fire protection program meets NRC and other requirements.

q. Explain fire protection requirements to the licensee's staff when required.

According to the licensee's staff and Station Nuclear Engineering Department (SNED) procedure number PE Q.44, a qualified corporate fire protection engineer reviews new plant modifications prior to implementation by the Architect-Engineering firm. This appears to be the extent of the corporate fire protection engineer's involvement. The qualifications of the Station Fire Marshal do not appear to be commensurate with the list of responsibilities assigned to that position. This lengthy list of responsibilities constitute a workload that may not be achievable by a single individual, regardless of the individual's qualification and experience.

To resolve this concern, the licensee is requested to provide at the site, a written evaluation (complete work study) of the responsibilities assigned to the station fire marshal. This evaluation should make a determination of the fire marshal's ability to effectively achieve each delegated responsibility based on his qualifications and time constraints.

This is considered an Unresolved Item (237/85033-02; 249/85029-02(DRS)) pending Region III's review of this evaluation.

4. Technical Specification Surveillance Review

Technical Specification 6.2.A.11 requires that detailed written procedures be developed, approved and adhered to for implementation of the Fire Protection Program. The inspector's review of the licensee's surveillance procedures and test results for fire protection Technical Specification surveillance requirements resulted in identification of the following discrepancies:

a. Testing of Diesel Fire Pump at Least Once Per Operating Cycle

Section 4.12.B.1.(e) of Technical Specification No. 3.12.B requires that the station diesel fire pumps be demonstrated operable by performing a system functional test which includes simulated automatic actuation of the pumps throughout their operating sequence. The licensee's commitment in Section 3.5.E.2 of the Fire Hazard Analysis Report dated April, 1977, requires the fire pump installations to conform to NFPA standard No. 20. This commitment states that a plant modification would provide an adequate flow gage for full flow testing of the pumps in accordance with NFPA standard 20. The licensee's surveillance procedure Nos. DFPP 4124-3 and DFPP 4124-4 were deficient in that:

(1) The procedure required manual throttling of the pumps to achieve the specific flows contained in Technical Specification 3.12.B. and did not address automatic activation.

(2) The procedures required testing the pumps to the specific head and flow contained in the Technical Specification No. 3.12.B,

but failed to require testing for head and flow as specified in NFPA 20.

(3) Measurement of quality affecting parameters such as pump vibration under full flow conditions were not included in the test procedure or the test results.

(4) The test results were not compared to the original manufacturer's shop test curve or field acceptance test for the pumps because neither of these curves were available to the licensee's staff.

b. <u>Testing of Water Suppression Systems at least Once Per Operating Cycle</u>

Section 4.12.B.1.(e) of Technical Specification No. 3.12.B requires that fire suppression water systems be demonstrated operable by performing a system functional test which includes simulated automatic actuation of the systems throughout their operating sequence. The licensee's commitment in Section 3.5.E.3 of the Fire Hazard Analysis Report requires that automatic sprinkler systems conform to NFPA Standard No. 13.

The licensee's surveillance procedure No. SP 84-6-39 failed to incorporate appropriate test requirements to demonstrate the sprinkler system is operable in accordance with NFPA 13 in that:

(1) The procedure did not require flow from the inspector's test valve of wet sprinkler systems. Instead, the alarm bypass valve was used for this test.

(2) The procedure did not require flow from the two inch drain valve of wet or dry systems. Instead, the alarm bypass valve was used for this test.

c. <u>Semiannual Testing of Fire Detectors</u>

Section 4.12.A of Technical Specification No. 3.12.A requires that the fire detection system be demonstrated operable by performing a channel functional test every six months. The licensee commitment in Section 3.5.E.1 of the Fire Hazard Analysis Report requires that the fire detector system conform to the requirements of NFPA Standard 72D.

The licensee's surveillance procedure No. DFPP 4185-2 (Revision 4) failed to incorporate the following quality affecting parameters as required by NFPA 72D:

(1) Periodic cleaning of detector units.

(2) Periodic adjustment for sensitivity (Section 3.1.2 of the original SER required this test to be conducted).

According to the licensee's staff, an independent fire protection consultant has been employed to review all technical specification procedures and test results to evaluate their adequacy in accordance with NFPA standards and design requirements. This assessment was in progress at the time of the inspection and is expected to be completed by the end of 1985. According to the licensee, where necessary, the procedures will be revised to coincide with the governing code and design requirements.

This is considered an Open Item (237/85033-03; 249/85-029-03(DRS)) pending Region III's review of the licensee's actions.

No violations or deviations were identified.

5. Administrative Controls - Control of Welding, Cutting, and Ignition Sources

Licensee procedure No. DAP 3-11 (Revision 4) contained what appears to be acceptable instructions for controlling storage of flammable and combustible liquids, storage of compressed gas cylinders, and accumulation of rubbish and transient combustibles such as wood scaffolding, etc. The procedure specifies housekeeping and cleaning responsibilities to be followed by all employees and contractors.

No violations or deviations were identified in this area, however; the inspectors cautioned the licensee on a proposed revision to welding and cutting procedure No. DMP 4100-1 that would include a provision to facilitate ALARA concerns in high radiation areas. The inspectors informed the licensee that any relief from the requirements for a firewatch to remain in the immediate area thirty minutes after cutting and welding has been completed would have to be discussed with NRR.

6. Quality Assurance Program

The licensee's commitment to Quality Assurance for fire protection is documented in Section 3.3 of "Information Relevant to Fire Protection Systems and Programs" and in letters to the NRC on this subject dated January 24, February 24, March 20, and March 27, 1978, January 31 and April 30, 1979.

The inspectors review of the licensee's Quality Assurance Program for Fire Protection included review of the following:

a. Eleven criteria applicable to fire protection that satisfy Appendix A to Branch Technical Position 9.5-1 and supplement guidance "Nuclear Plant Functional Responsibilities, Administrative Controls and Quality Assurance."

b. Quality Assurance Surveillance Reports dated September 3-6, 1985, September 5-9, 1985, September 9-13, 1985, and September 16-30, 1985.

c. Annual Quality Assurance Audits Nos. QAA 12-84-I dated April 17, 1984, and QAA 12-83-I dated April 15, 1983.

d. Triennial Audit by M&M Protection Consultants dated December 4, 1984.

No violations or deviations were identified; however, the inspectors suggested to the licensee that for clarification, the statements made in Section 3.3 of the "Information Relevant to Fire Protection Systems and Programs" should be modified to indicate their specific commitment to a QA program to fire protection. As written, this statement can be interpreted to mean that the licensee committed to apply all of the criteria of Appendix B in 10 CFR 50 to fire protection.

The inspectors determined that the licensee's practice of considering fire protection as reliability-related is acceptable because this practice ensures that all of the eleven criteria contained in the NRC's Guidance are included in the program. In addition, this practice allows for the normal QA program for safety-related systems to be applied to fire protection in it's entirety. Only one QA manual exists for reliability-related systems and fire protection systems.

Although the licensee's Quality Assurance Program appears to be effectively identifying issues that are contributing to hardware and programmatic weaknesses, the licensee does not appear to be taking prompt and effective corrective actions. This is exemplified by the remaining open items that have been identified in QA audits and surveillances, LERs, and NRC inspections. (This is further discussed in 3.b.(5) of the report.)

7. Plant Tours

During tours of the plant, the inspectors observed the following deficient conditions:

a. Failure to Comply with License Condition No. 2.B. of Amendment No. 33 to Facility Operating License No. DPR-25 and Amendment No. 36 to Provisional Operating License No. DPR-19.

Section 5.1.6.6 of the original Fire Protection SER for Dresden Units 2/3 dated March 22, 1978 states that the licensee proposed the installation of an automatic fire detection system to provide early warning of a fire in the Refueling Floor Area in order to satisfy the objectives of Criterion 3 of Appendix A to 10 CRF 50. Amendment No. 36 to Provisional Operating License No. DPR-19 (Unit 2) and Amendment No. 33 to Facility Operating License No. DPR-25 (Unit 3) dated October 1, 1980, require that the early warning automatic fire detection system for the refueling floor area be installed by start up following the 1979 Unit 3 refueling outage.

As of the date of this inspection (approximately six years after start up following the Unit 3 1979 refueling outage) the licensee has failed to comply with the provisions of Amendment No. 36 to Provisional Operating License No. DPR-19 and Amendment No. 33 to Facility Operating License No. DPR-25. An early warning automatic fire detection system fire detection system has not been installed

in the Refueling Floor Area and no compensatory measures have been taken as a result of this decreased effectiveness of the plant's fire protection features.

The installation of an automatic early warning fire detection system in the refueling floor area was not discussed in any of the licensee's correspondence to the NRC that requested amendments to modify the plant's fire protection Technical Specifications to incorporate Limiting Conditions for Operation and Surveillance Requirements for the fire protection modifications required by the original SER for Dresden Units 2/3. None of the proposed Tables 3.12.1 to Technical Specification 3.12 listed fire detection instruments in the refueling floor area. However, sufficient information existed which should have alerted the licensee that he was in violation of a license condition. For example:

(1) By letter dated February 25, 1980 (R. F. Janecek-CECO to T. A. Ippolito-NRC) the licensee noted that they did not believe installation of an automatic early warning fire detection system in the refueling floor area was warranted based on low fire loading and the ability to make up water and cool the spent fuel pools in the event of a loss of either Unit's spent Fuel pool cooling equipment due to fire. This letter did not request relief from the installation of a refueling floor fire detection system. No official NRC response was issued for this letter.

(2) By letter dated March 18, 1980 (L. Derderian-NRC to M. Antonetti - Gage Babcock and Associates - Consultants to the Licensee) the NRC referenced a March 17, 1980 telecon record with T. Pickens (CECO) in which the following was agreed to concerning Reactor Building Refueling floor fire detection systems for Dresden Units 2/3 and Quads Cities Units 1 and 2:

   (a) The license was to confirm to the NRC that in the most heavy fire loading situations (i.e. refueling periods), the loading would not exceed that necessary to cause structural failures.

   (b) The licensee was to confirm that structural concrete protection extends from the floor to some specified height, lessening the likelihood of structural failure.

   (c) The licensee was to recalculate average combustible loading subtracting out the pool areas.

   The licensee could not provide the inspectors with documented evidence that these issues were addressed.

This failure to followup on implementation of a license condition is indicative of a programmatic breakdown which has resulted in a reduced level of fire protection than was intended to satisfy

criterion 3 of Appendix A to 10 CFR 50 and is considered a violation of Amendment No. 36 to Provisional Operating License No. DPR-19, Amendment No. 33 to Facility Operating License No. DPR-25, 10 CFR 50 (237/85-033-04; 249/85-029-04)(DRS).

b. <u>Preparations for the Upcoming Extended Unit 3 Outage Separation of Unit 1 from Units 2/3</u>

During plant tours and in meetings with the licensee during the inspection, the licensee agreed to update their response to the NRC and describe the administrative controls and the actions that will be necessary to isolate Unit 1 from Units 2 and 3 since Unit 1 is no longer operational but shares common areas with Units 2 and 3. The inspectors also requested that the licensee describe those administrative controls and actions that will be necessary to separate common areas in Units 2/3 while Unit 2 is operating and Unit 3 is in an extended outage.

This is considered an Open Item (237/85-033-05; 249/85-029-05)(DRS) pending further review by Region III.

c. <u>Self Contained Breathing Air Supply for the Fire Brigade</u>

Section 3.4.D.4(h) of the document entitled "Information Relevant to Fire Protection Systems and Programs", requires that breathing apparatus using full face piece positive pressure masks that are approved by NIOSH be provided for the fire brigade.

The inspectors examined the fire brigade Scott Air Pak breathing air cylinders that were provided on Fire Chart No. 2. Four out of four of these cylinders contained 1800 pounds of air pressure. According to the licensee's staff, a minimum of 2200 pounds of air pressure should be contained in each cylinder. 2400 pounds of air pressure would indicate the cylinder is full and may provide a 30 minute air supply for the average fire brigade member. The cylinder gauges have a range of up to 3000 pounds of air pressure.

A December 1984 three year audit recommended that a set of written instructions be provided at the breathing air cylinder filling station to assure that the cylinders are filled properly. Filling of the cylinders is the responsibility of Health Physics. Due to time constraints, the inspectors were unable to contact Health Physics to follow up this concern. Therefore, the licensee is requested to provide at the site the appropriate acceptance criteria for filling breathing air supply cylinders. This is considered an Open Item (237/85-033-06; 249/85-049-06)(DRS) pending Region III review of the licensee's breathing air cylinder filling procedures.

d. <u>300 Pound Fixed Cardox System Supply Tank First Floor, Turbine Building</u>

During plant tours, the inspectors observed the following deficiencies on the main CO_2 system storage tank located on the first floor of the turbine buildings.

(1) The access door to the tank compressor motor was missing.

(2) The glass cover to the tank's mercoid switch located inside the access door was missing.

The licensee had no explanation for these deficiencies, but agreed to take immediate corrective actions.

This is considered an Open Item (237/85-033-07; 249/83-029-07)(DRS) pending further verification of the licensee's corrective actions by Region III.

8. Open Items

Open items are matters which have been discussed with the licensee, which will be reviewed further by the inspector, and which involve some action on the part of the NRC of licensee of both. Open items disclosed during the inspection are discussed in Paragraphs 4.c, 5.a, 7.b, 7.c, 7.d.

9. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, items of noncompliance, or deviations. An unresolved item disclosed during the inspection is discussed in Paragraph 3.c.

10. Exit Interview

The inspectors met with the licensee representatives at the conclusion of the inspection on October 4, 1985, and summarized the scope and findings of the inspection. The licensee acknowledged the statements made by the inspectors. The inspectors also discussed the likely informational content of the inspection report with regard to documents reviewed by the inspector during the inspection. The licensee did not identify any such documents as proprietary. On October 21, 1985, in a telephone conversation with the licensee additional concerns regarding the lack of fire detectors on the refueling floor were discussed with the licensee.

Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690

January 24, 1986

Mr. James G. Keppler
Regional Administrator
U.S. Nuclear Regulatory Commission
Region III
799 Roosevelt Road
Glen Ellyn, IL 60137

Subject: Dresden Station Units 2 and 3
Response to Inspection Report Nos.
50-237/85-033 and 50-249/85-029
NRC Docket Nos. 50-237 and 50-249

Reference: Letter from J. G. Keppler to Cordell Reed
dated December 26, 1985.

Dear Mr. Keppler:

This letter is in response to the inspection conducted by Messrs. J. Holmes and C. Ramsey of your staff between September 30 and October 21, 1985, of activities at Dresden Station. The referenced letter indicated that certain activities appeared to be in noncompliance with NRC requirements. The Commonwealth Edison Company response to the Notice of Violation is provided in the enclosure.

In addition to the response to the Notice of Violation which we have provided, we have also attached our current plans for resolving the remaining concerns that the inspector identified in his report. These plans are described in Attachment B.

If you have any further questions on this matter, please direct them to this office.

Very truly yours,

D. L. Farrar
Director of Nuclear Licensing

lm

Attachment

cc: NRC Resident Inspector - Dresden
1171K

ATTACHMENT A

COMMONWEALTH EDISON COMPANY

RESPONSE TO NOTICE OF VIOLATION

DESCRIPTION OF VIOLATION

10 CFR 50.48 (a) requires that each operating nuclear power plant have a fire protection plan that satisfies Criterion 3 of Appendix A to 10 CFR Part 50. It further requires that the plan shall describe specific features necessary to implement the program such as administrative controls and personnel requirements to limit fire damage to structures, systems, or components important to safety so that the capability to safely shut down the plant is ensured.

Section 3.1.A.1 of the licensee's Fire Hazards Analysis Submittal, which forms part of the licensee's approved fire protection program, states that the licensee has a Fire Protection Coordinator whose responsibilities include, in part, program coordination, equipment procurement, program enhancement, conducting inspections, and supervising training of personnel.

Contrary to the above, the licensee has failed to consistently and effectively staff the Fire Protection Coordinator position with the result that certain fire protection equipment was not installed, hardware and equipment were not being properly maintained, required training was not completed, and prompt and effective corrective action was not taken for identified deficiencies.

DISCUSSION OF RESPONSE TO THE VIOLATION

At the time Section 3.1.A.1 of the Fire Hazards Analysis was written, a Fire Protection Coordinator reported to the System Safety Department. Subsequently, the Fire Protection Coordinator was transferred to the Quality Assurance Department. Shortly thereafter, the individual filling this position retired. Currently, the Company employs three Fire Protection Engineers in the General Office. Two of these Fire Protection Engineers are in the Nuclear Services Technical Department, the third is in the Quality Assurance Department and has the title of QA Fire Protection Coordinator. Many of the Fire Protection Coordinator's duties listed in Section 3.1.A.1 are currently performed by the QA Fire Protection Coordinator, NST Fire Protection Engineers and Station personnel. Thus, subsequent to the initial submittal of Section 3.1.A.1, the Company has employed three Fire Protection Engineers in the General Office in order to improve the fire protection program.

CORRECTIVE ACTION TAKEN AND THE RESULTS ACHIEVED

An NST Fire Protection Engineer is now at Dresden approximately one day per week to assist the Station. This person will continue in this capacity until the Task Force report, which is discussed below, is accepted and implemented. It is expected that the Task Force will recommend a course of action that will relieve NST from the weekly requirement.

CORRECTIVE ACTION TO BE TAKEN TO AVOID FURTHER VIOLATIONS

A task force has been assembled to examine the various fire protection duties and tasks that have to be performed on a company wide basis. The task force has been instructed to report their recommendations for improvements in the fire protection program, including organizational and staffing requirements, to the Vice President of Nuclear Operations by April, 1986.

DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED

Full compliance will be achieved at such time as the task force recommendations have been reviewed, evaluated and implemented to the extent deemed necessary. We will provide a follow-up response addressing the task force recommendations by July 1, 1986.

ATTACHMENT B

COMMONWEALTH EDISON COMPANY

PLANS FOR RESOLVING FIRE PROTECTION ISSUES

This attachment responds to the issues identified in the routine safety inspection conducted by Messrs. J. Holmes and C. Ramsey at Dresden Nuclear Power Station on September 30 through October 21, 1985. Many of the items identified by the inspectors as examples of programmatic breakdowns had already been identified during a review of the regulatory, commitment, and code compliance in the fire protection area at our operating stations. We feel that the review which we had undertaken has demonstrated our commitment to ensuring that all fire protection features at our stations have been implemented.

Our present expectations for addressing those items identified by the inspector as indicative of a programmatic breakdown are as follows:

A. The fire detection system not installed on the refueling floor (Paragraph 7.a of the inspection report) was identified as part of the Company's Appendix R reassessment project. Since the SER items were presented previously at the enforcement conference, no further response is required at this time.

B. Maintenance of fire protection equipment and hardware was corrected as follows:

(1) Work is in progress to chain and lock the hose station root valves. (Paragraph 2d of the inspection report) We expect that the valves will be locked and procedure revisions will be completed by August 31, 1986.

(2) A modification was initiated in 1984 to install fire detection and sprinkler system alarms in accordance with NFPA 72 D. (Paragraph 2e of the inspection report). This modification and related surveillance procedures will be completed and placed in service in sections. All portions of the modification are presently scheduled to be complete by the end of the Fall 1986 Unit 2 Refueling outage.

(3) Items identified by the inspector's review of Technical specification surveillances (Paragraph 4 of the inspection report) are being resolved as follows.

a.) Diesel Fire Pump surveillance procedures are in the process of being revised as a result of our NFPA Code Review. These revised procedures are expected to be implemented by August 31, 1986.

b.) Water suppression system surveillance procedures and piping changes are in progress as a result of our NFPA Code Review. These revised procedures and necessary piping changes are expected to be implemented by August 31, 1986.

c.) A modification is in progress on our fire detection system, and surveillance procedures are being revised in accordance with NFPA 72D as a result of our NFPA Code review. This modification and related surveillance procedure will be completed and placed in service in sections. All portions of the modification are scheduled to be complete by the end of the Fall 1986 Unit 2 Refueling outage.

(4) In the area of Administrative Controls (Paragraph 5 in the report) the inspectors cautioned the licensee on a proposed revision to welding and cutting procedure DMP 4100-1 that would include a provision to facilitate ALARA concerns in high radiation areas.

DMP 4100-1 will be revised to clearly require the 30 minute fire watch within line-of-sight of the work area. This procedure was in the process of revision as a result of our NFPA Code Review. This revision is presently scheduled for completion by March 14, 1986.

(5) During tours of the plant (Paragraph 7 of the inspectors report) the inspector identified deficient conditions which are being corrected as follows:

a.) The inspector identified the lack of refueling floor detection as a violation. The violation notice indicated no further response is necessary for this item.

b.) The inspector raised concerns about isolation of Unit 1 from Units 2 and 3, and administrative controls and actions necessary to separate common areas in Units 2/3 while Unit 2 is operating and Unit 3 is in an extended outage. A stricter transient combustible control procedure is being developed, and is presently scheduled for implementation by September 30, 1986. A cognizant foreman has been designated to assist the fire marshal in timely correction of housekeeping deficiencies. The Unit 3 Recirculation Piping Replacement primarily involves the drywell of the shutdown unit and does not affect common fire barriers. However, a detailed memorandum discussing the proper handling of fire barriers has been discussed with all personnel at the Station as part of the weekly "tailgate" staff meetings. Also procedure DFPP 4175 -1, Fire Barrier

Integrity and Maintenance, has been revised to further clarify the proper handling and maintenance of fire barriers, including fire doors, fire walls, penetration seals for mechanical and electrical components, and fire dampers. The separation of Unit 1 is being covered by the Appendix R review program. This information is being added to the updated Fire Hazards Analysis for Units 2 and 3.

c.) A procedure is being developed by the Radiation/ Chemistry Department which will provide standards for the proper refilling of the SCBA air packs. This procedure will be posted at the air pack refilling station. The procedure is presently scheduled to be implemented by June 30, 1986.

d.) The missing door and glass cover have been replaced on the carbon dioxide system storage tank.

C. The inspector identified technical specification surveillance procedures that did not incorporate appropriate testing of quality-affecting parameters in accordance with design and governing code requirements. (Paragraph 4 of the inspectors report). Our resolution to items in Paragraph 4 of the inspectors report is discussed above.

D. The inspection report states that administrative controls did not adequately control fire protection features as discussed in Paragraph 5 of the report. As indicated in our above response to paragraph 5, the welding and cutting procedure is being revised to resolve the inspector's concern.

E. The inspection report stated that many deficiencies identified in LERs, NRC inspections, QA audits and QA surveillances did not receive prompt or effective corrective action. These items are identified in Paragraphs 2 and 6 of the report. Their resolution is as follows.

(1) The long term corrective actions have been completed for the Auxiliary Electric Equipment Room HVAC dampers. (Paragraph 2a of the report)

(2) Paragraph 2d discusses hose station root valves. Our resolution is discussed above.

(3) Paragraph 2e discusses the interconnection of the security system computer with the plant fire detection and sprinkler system alarms. Our resolution is addressed by the proposed implementation of the 1984 modification to install fire detection and sprinkler alarms for NFPA 72D.

(4) Paragraph 21 discusses deficiencies in portable fire extinguishers. A dedicated work crew has been established to eliminate the backlog of fire protection work requests. As of January 8, 1986, this backlog has been eliminated. The fire extinguisher discrepancies are tentatively scheduled for completion of corrective action by January 31, 1986.

(5) Paragraph 2m discusses fire brigade drills and training. An assessment will be made of fire brigade drills, training, and practice sessions, and the three-year independent critiques of fire brigade drills. The assessment is presently scheduled to be completed by August 31, 1986.

(6) Paragraph 2n discusses Pre-fire plans. Pre-fire plans have been developed and are in the process of being implemented. Full implementation is expected by March 14, 1986.

(7) Paragraph 2o discusses hands-on fire brigade training. As stated above, an assessment will be made of fire brigade training.

(8) Paragraph 6 discusses the apparent lack of prompt and effective corrective actions to problems identified by the QA program. As discussed above, a dedicated work crew has been established to eliminate the backlog of fire protection work requests.

F. The inspection report identifies weaknesses in the scheduling of fire drills. (Paragraphs 2m and 2o of the report) As discussed above, an assessment of the brigade training program will be made.

1171K

July 7, 1986

DJS LTR: 86-477

TO: J. R. Wojnarowski

SUBJECT: Review of Commitments Made in Dresden Station Units 2 and 3 Response to Inspection Reports No. 50-237/85-033 and 50-249/85-029

REFERENCES:
1) Letter of January 24, 1986 from D. L. Farrar to J. G. Keppler, Response to Notice of Violation (NL-86-0131).
2) Letter from J. G. Keppler to Cordell Reed, dated December 26, 1985.

As you requested by phone July 1, 1986, the commitments associated with the above-referenced letters have been reviewed. The attached table provides a status update regarding the Dresden Action Items. If there are any questions, please contact R. Whalen at extension 665.

Prepared by ______________________
R. Whalen
Technical Staff

Approved by ______________________
D. J. Scott
Station Manager
Dresden Nuclear Power Station

DJS:RW:hjb
Enclosure
cc: J. Achterberg
M. Dillon
R. Christensen
R. Whalen
B. Zank
D. Adam
B. Rybak
G. Smith
J. McDonald
T. Hausheer
R. Hunnicutt
S. Becker
File/Fire Protection
File/Numerical

FIRE PROTECTION AUDIT ACTION ITEMS

Action Item Per Attachment B, Reference 1	Description	Commitment Date	Current Status	Cognizant Person
E.4	Fire extinguisher discrepancies.	01/31/86	Completed on schedule.	M. Dillon
B.4	Revising DMP 4100-1, cutting and welding procedure to insure continuous fire watch 30 minutes after work stops.	03/14/86	Complete. Procedure was approved 2/28/86.	B. Geier
E.6	Implementation of pre-fire plans.	03/14/86	Complete. On-Site Reivew was completed 3/13/86.	M. Dillon
B.5.C.	Posted procedure for refilling SCBA air packs.	06/30/86	Complete. Procedure DRP 1310-11 was approved 6/2/86.	L. Burczak
B.1	Chain and lock hose station root valves; change valve checklist as appropriate.	08/31/86	Locks have been purchased and work is proceeding on schedule.	M. Dillon
B.3.a.	Revise diesel fire pump surveillance procedures to meet NFPA 20 requirements.	08/31/86	Complete. DFPP 4123-6 (2/3 Diesel Fire Pump Annual Capacity Test) and DFPP 4123-7 (Unit 1 Diesel Fire Pump Annual Capacity Test) were revised to include acceptance curves 6/30/86. These revisions incorporate items 4.a.(1) through (4) as listed in Enclosure 3 of Reference 2, with the exception that automatic activation testing is covered under operability surveillances DFPP 4123-5 and DFPP 4123-1.	R. Whalen

Action Item Per Attachment B, Reference 1	Description	Commitment Date	Current Status	Cognizant Person
B.3.b.	Revising suppression system surveillance tests to meet NFPA 13 requirements.	03/31/86	DFPP 4114-2 and 4114-3, Reactor and Turbine Building Monthly Fire Equipment Inspection, will be revised to include a waterflow alarm check on the west pipe systems from the remote inspector's test location. This requires completion of certain modifications, some of which may not be completed until after 8/31/86. An evaluation of this approach is being performed by a qualified fire protection engineer to insure that the requirements are met.	R. Hunnicutt/ T. Hausheer
			Note: Section 4.b in Enclosure 3 of Reference 2 also refers to Technical Specification 4.12.B.1.(e), which addresses a triennial flow test of the underground mains. DFPP 4123-8 was approved for use 6/30/86, and will be used in place of SP-84-6-39. However, the inspector's concerns about alarm testing do not appear to apply in this case.	R. Whalen
E.5	Fire Brigade drills and training assessment.	08/31/86	Regarding the frequency of Fire Brigade drills, it is believed that Dresden is committed only to the following position from an August 8, 1977 letter from M. Turbak (NLA) to Davis (NRC).	M. Dillon/ T. Hausheer/ S. Becker

Action Item Per Attachment B, Reference 1	Description	Commitment Date	Current Status	Cognizant Person
E.5 - (Cont'd)			"Fire Drills are conducted monthly in accordance with approved station surveillance schedules. The designation of which shift will conduct a specific drill is the responsibility of the Fire Marshal. When a fire drill is conducted by NML the Fire Brigade Leader (Fire Chief) as well as the Fire Brigade, are evaluated..." Note: Currently the Fire Brigade Leader is evaluated during <u>all</u> drills. In process of issuing the December 2, 1980 fire protection SER supplement the NRC seems to have accepted the existing drill program since it specifically references the August 8, 1977 Turbak letter as having been reviewed. However, an assessment is being performed of this position. Regarding the hands-on Fire Brigade training, implementation plans are under review by the Training Department. An implementation plan is scheduled for development by 8/31/86, including a timeline for resolving this item.	

Action Item Per Attachment B, Reference 1	Description	Commitment Date	Current Status	Cognizant Person
B.5.b	Control of transient combustibles.	09/30/86	A transient combustible procedure is being developed. and is scheduled for implementation by 9/30/86. Also, the Unit 1 separation concerns are being incorporated into the Unit 2/3 fire hazards analysis.	R. Whalen
B.2, B.3.c.	Installation of detection/ alarm system separate from the security system and addressing cleaning/ sensitivity testing issues.	03/01/87 (End of U-2 outage)	Modification work is proceeding on schedule.	R. Hunnicutt

Tab 4

DRESDEN 2 & 3

FIRE PROTECTION PROGRAM DOCUMENTATION PACKAGE

Inspection Report No. 50-249/86006

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

FEB 26 1986

Docket No. 50-249

Commonwealth Edison Company
ATTN: Mr. Cordell Reed
Vice President
Post Office Box 767
Chicago, IL 60690

86 696
REC'D FEB 27 '86

Gentlemen:

This refers to the special safety inspection conducted by NRC Personnel of this office on January 28, 29 and February 7 and 13, 1986, of circumstances associated with a fire in the Dresden Nuclear Power Station, Unit 3 drywell expansion gap on January 20, 1986, and to the discussion of our findings with Mr. D. Scott at the conclusion of the inspection.

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.

No violations of NRC requirements were identified during the course of this inspection; however, the you are requested to formally respond to each of the issues identified in Paragraph 3 prior to Unit 3 restart.

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

Sincerely,

Carl J. Paperiello

Carl J. Paperiello, Director
Division of Reactor Safety

Enclosure: Inspection Report
No. 50-249/86006(DRSS)

cc w/enclosure:
D. L. Farrar, Director
of Nuclear Licensing
D. J. Scott, Plant Manager
DCS/RSB (RIDS)
Licensing Fee Management Branch
Resident Inspector, RIII
Phyllis Dunton, Attorney
General's Office, Environmental
Control Division

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-249/86006(DRSS)

Docket No. 50-249 License No. DPR-25

Licensee: Commonwealth Edison Company
P.O. Box 767
Chicago, IL 60690

Facility Name: Dresden Nuclear Power Station, Unit 3

Inspection Location: Morris, IL

Inspection Conducted: January 28, 29 and February 7 and 13, 1986

Inspectors: C. Ramsey — W. Guldemond for — 2/25/86 Date

J. Patterson — for — 2/26/86 Date

Approved By: W. Guldemond, Chief, Operational Programs Section — 2/25/86 Date

Inspection Summary

Inspecton on January 28, 29 and February 7 and 13, 1986 (Report No. 50-249/86006(DRS))
Areas Inspected: Announced special safety inspection conducted to review potential damage to the facility originating from a fire in the drywell expansion gap on January 20, 1986. The inspection involved 60 inspector-hours by six NRC inspectors.
Results: No violations or deviations were identified.

DETAILS

1. Persons Contacted

CECo

D. Scott, Station Manager
R. Flessner, Services Superintendent
J. Brunner, Assistant Services Superintendent
T. Hauser, Fire Protection Engineer
R. Mirochina, SNED
D. Wilgus, SNED
M. Dillion, Station Fire Marshal
J. Schrange, Health Physicist

Rolf Jensen and Associates

J. Klien, Consultant

NRC

E. Hare, Resident Inspector
S. Stasek, Resident Inspector
L. McGregor, Senior Resident Inspector
R. Landsman, Region III Project Manager

2. January 20, 1986 Drywell Expansion Gap Fire

a. Apparent Origin of the Fire

At approximately 0830 hours on January 20, 1986, with Unit 3 shutdown and defueled, an air arc cutting activity began on containment pipe penetration No. 113 ("B" reactor water cleanup system pipe) inside the reactor water cleanup system (RWCU) "B" heat exchanger room. At 0905 hours workers in the area observed smoke in the vicinity of the pipe penetration. The shift engineer's office and the control room were notified at approximately 0916 hours.

b. Initial Response

The fire watch for the air arc cutting activity apparently discharged a dry chemical extinguisher on or in the vicinity of pipe penetration No. 113. Subsequently, a fire brigade leader arrived to investigate the fire and determined that the fire had been extinguished.

At approximately 1000 hours, the station fire marshal was notified by the shift engineer of smoke in the Unit 3 reactor building. At 1004 hours the reactor building ventilation system, which had been turned off to support Standby Gas Treatment System testing, was turned on to remove smoke from the Unit 3 reactor building and drywell, all personnel were evacuated from the Unit 3 torus and drywell areas, and air samples were taken to verify the quality of air for personnel safety.

At 1030 hours, the shift engineer contacted the station fire marshal and informed him that the smoke was clearing from the Unit 3 reactor building. Apparently the fire brigade leader and station construction concluded that the problem was under control because the fire watch had earlier discharged a dry chemical extinguisher on or in the vicinity of the pipe penetration in the RWCU heat exchanger room and smoke was being cleared from the reactor building and the drywell by the reactor building ventilation system.

At approximately 1120 hours personnel were allowed to reenter the drywell. At 1130 hours station technical staff personnel discovered a hot spot in the drywell in the vicinity of penetration No. 113. Workers complained of intense heat 4 to 5 feet away from the drywell steel liner. At 1155 hours, all personnel were again evacuated from the drywell because of the overheated drywell liner. A construction staff person took general use (not calibrated) pyrometer readings in the vicinity of pipe penetration No. 113 on the inside of the drywell liner (unexposed side) between 1230 and 1315 hours. The highest reading recorded was 440-450° F.

The heated drywell liner condition alerted the station fire marshal to investigate what could be burning on the other side of the drywell liner. His review of Section 5 of the Dresden FSAR identified the presence of polyurethane foam installed inside the drywell expansion gap between the steel liner and the concrete shell.

c. Drywell Expansion Gap Design

The outer surface of the steel drywell liner is enclosed in 8 feet of structural and shielding concrete. Thermal expansion of the drywell liner as a result of normal reactor operations will cause the liner to expand both radially and vertically. To accommodate this expansion, during construction, an expansion gap was provided between the structural concrete and the drywell steel liner. The sizing of the expansion gap was based on the maximum drywell steel liner temperature following a postulated loss of coolant accident.

d. Materials Used to Fill the Drywell Expansion Gap

To maintain sufficient space for liner expansion, prefabricated polyurethane foam sheets were installed over the entire liner exterior surface. Epoxy impregnated fiberglass tape was applied over all joints in the foam and one-fourth inch to 3/8 inch fiberglass-epoxy prefabricated cover panels were installed over the foam panels. The fiberglass panels were made of fibrous glass in chopped fiber form with an isophatallic resin as a binder.

Tests were conducted at the site on mockups of the drywell steel liner/polyurethane foam/fiberglass panels to determine their displacement from the pour of structural and shielding concrete. The test results showed that the fiberglass was displaced less than 1/4 inch from the pouring and curing of the concrete. Therefore, it was assumed that the drywell expansion gap design space was maintained during construction.

e. <u>Determination Made that Polyurethane Foam Panels Were Burning Inside the Drywell Expansion Gap</u>

As a result of the station fire marshal's review of Section 5.2.3.6 of the FSAR, he determined that hot slag (molten metal) from the air arc cutting activity on pipe penetration No. 113 in the RWCU heat exchanger room had come in contact with and ignited the polyurethane foam material in the drywell expansion gap. The typical drywell pipe penetration detail (figure 5.2.3.27 of the FSAR) shows a 2 inch gap between the pipe sleeve and the penetration sleeve, which provides a direct path to the polyurethane foam material. Furthermore, the drywell expansion gap is not air tight. The fiberglass panels installed over the polyurethane foam material do not form a barrier that will exclude air from coming in contact with the polyurethane foam material. The 45 degree angle that pipe penetration No. 113 is installed through the drywell adds credibility to this hypothesis as to the origin of the fire.

f. <u>Extinguishment of the Fire</u>

Since the fire was determined to be in a concealed space that was impossible for the fire brigade to reach, the station fire marshal directed the fire brigade leader to start applying water from a 1 1/4 inch (3/4 inch inside diameter) rubber hose (supplied by the demineralized water system at 100 PSI) to the 2 inch gap between the pipe sleeve and the penetration sleeve on penetration No. 113. This action was initiated between 1230 and 1300 hours. As the fire marshal was not certain that water applied through this penetration would extinguish the fire, additional hose streams supplied by the fire water system at 100 PSI were applied above and adjacent to the penetration (pipe penetration Nos. 133, 122, 144 and 143).

At 1330 hours, the licensee decided to monitor the drywell liner temperature on the inside of the drywell. At 1700 hours, inside drywell liner temperatures were recorded at 140, 110 and 90° F. At 1730 hours, the licensee's corporate fire protection engineers and the station fire marshal considered the fire to be extinguished due to declining inside drywell liner temperatures. At 2100 hours, inside drywell liner temperatures were determined to be normal and the application of water to the drywell expansion gap was discontinued. No offsite fire department assistance was requested and no emergency event was declared by the licensee at any point during this event.

g. <u>Potential Damage Resulting From the Fire</u>

At the time of the inspection the licensee had not determined the extent of damage resulting from the fire. In two principal areas inside the drywell (approximately 10 feet in diameter and 10 feet apart), charred, discolored, blistered or burned away paint was visible on the drywell liner. The drywell steel liner is approximately 1 1/8 inch thick carbon steel.

Polyurethane foam materials are synthetically produced from glycols and diisocyanates. It has been established by actual fires and certified fire testing laboratories that urethane foam materials ignite easily and burn vigorously with the production of dense black smoke and a very black, viscous melt product which can burn with the intensity of a flammable liquid (Reference Underwriters Laboratories Inc. and Factory Mutual Laboratories Inc. 1969-74 studies on the Flammability of Cellular Plastics). Burning polyurethane materials also produce corrosive and toxic oxides of nitrogen, together with other toxic gases and corrosive that are harmful to metals.

It appears that this fire began some time after 0830 hours, when the air arc cutting activity began on pipe penetration No. 113. It burned with some intensity and it is suspected that high temperatures were reached inside the drywell expansion gap. It is not known how much polyurethane foam material was consumed by the fire or how far the fire spread vertically or horizontally around the drywell. The 4½ hours burn time from 0830 hours to 1300 hours (when water was first applied through penetration No. 113) indicate that substantial burning may have occurred.

Apparently, a substantial amount of water was applied to the drywell expansion gap to extinguish the fire (approximately 500 gallons per minute (GPM) for 8 hours or 240,000 gallons). However, according to the licensee, only 20,000 gallons of excess water was removed from the torus basement by the radwaste system the day after the fire.

The licensee did provide the inspectors with a draft copy of proposed work to be performed by Sargent and Lundy (S&L Project No. 7368-30) to evaluate the integrity of the Unit 3 drywell for affects from the fire. This evaluation did not appear to consider some of the specific NRC concerns detailed in Paragraph 3 of this report and is not scheduled to be completed until March 31, 1986.

h. Emergency Preparedness Implications

The inspector reviewed records associated with the event; interviewed several available persons knowledgeable of the event; and reviewed the Station's Emergency Action Levels (EALs) and the notification requirements of 10 CFR 50.72 for applicability. The event was not classifiable as an emergency per the current EALs for the Fire Condition (No. 5) for the following reasons: offsite fire fighting assistance was not requested; equipment was not degraded such that a Limiting Condition for Operation (LCO) required a reactor shutdown; equipment was not degraded such that a cold shutdown or hot shutdown could not be achieved or maintained; and required safety systems were not potentially affected. Since all fuel had been removed from the reactor vessel for some months, there was no need to be able to achieve and maintain shutdown and no reactor safety systems were required to be in operation. The event was not classifiable as an emergency per the current EALs for the "miscellaneous" Condition (No. 18) which was worded as follows: "any other conditions of equivalent magnitude to the criteria used to define the accident category as determined by the Station

Director." The Unusual Event EAL for Condtion No. 18 listed a number of circumstances that warranted increased awareness on the part of State and/or local offsite officials. The Alert EAL for Condition No. 18 listed several circumstances which warranted precautionary activation of the onsite Technical Support Center (TSC) and near site Emergency Operations Facility (EOF). The Site Area Emergency EAL for Condition No. 18 addressed activation of these Emergency Response Facilities, radiological monitoring teams, and precautionary notification of the public near the site. The General Emergency EAL for Condition No. 18 addressed an imminent core melt situation. No EAL associated with Condition No. 18 was applicable to the fire incident.

Since no EAL was applicable, an emergency declaration and activation of the Generating Stations Emergency Plan (GSEP) did not occur. Consequently, initial notifications of the Illinois Department of Nuclear Safety and Illinois Emergency Services and Disaster Agency were neither required nor performed. Similarly, initial notification of the NRC Operations center was not required per 10 CFR 50.72(a); however, the licensee did notify the Station's Senior Resident Inspector between 4 p.m. and 5 p.m. on January 20. That individual informed his supervisor. Neither the licensee nor the aforementioned Region III personnel deemed it necessary to promptly notify the NRC Operations Center per the requirements of 10 CFR 50.72(b) or (c). Due to the extensive nature of maintenance being performed on the Unit 3 reactor coolant system, and the fact that the vessel had been completely defueled for some months, regional emergency preparedness staff have also concluded that the requirements of 10 CFR 50.72(b) and (c) were not applicable to this situation.

The wording of the Unusual Event EAL for Condition No. 18 was not in close agreement with regulatory guidance found in NUREG 0654, Revision 1. The licensee's EAL stated, in part, that "a condition that warrants increased awareness on the part of the State and/or local offsite officials." Relevant regulatory guidance for the Unusual Event classification states, in part, that "other plant conditions exist that warrant increased awareness on the part of a plant operating staff (emphasis added) or State and/or local offsite authorities." During the course of the licensee's response to the fire incident, there were a number of meetings in the TSC involving Station management and/or technical staff; personnel were evacuated from the reactor building for a time; the licensee's General Office was informed of the incident; and personnel made repeated entries into the drywell to obtain temperature readings to help determine whether the fire still existed. There was clearly increased awareness and activity by plant operating and other plant staffs in response to the fire. Had the Unusual Event EAL for Condition No. 18 included the phrase "plant operating staff," per the regulatory guidance, there would be no question whether or not the NRC Operations Center and appropriate State agencies needed to be promptly informed of the fire incident, per the requirements of 10 CFR 50.72 and 10 CFR 50, Appendix E, Paragraph IV.D.3.

Therefore, to prevent recurrence of any uncertainties regarding the need for the licensee to promptly inform the NRC Operations Center and appropriate State agencies of significant responses by Station operations personnel to abnormal conditions onsite, the phrase "plant operating staff" should be added to the Unusual Event EAL for Condition No. 18.

3. <u>NRC Request For Information To Be Provided By The Licensee Prior To Unit 3 Startup From The Current Outage</u>

In view of the damage that may have occurred to the drywell steel liner, the structural and shielding concrete, electrical and pipe penetrations, or other structures and equipment required for safe operation of the Unit, the licensee is requested to provide to Region III a detailed assessment of this event that will include a confirmation of short term and long term operability of the affected structures, systems and/or components. This assessment must include an evaluation of the following concerns for Region III and Office of Nuclear Reactor Regulation review prior to restart of Unit 3 from the current outage:

a. <u>Detailed Chronology of the Fire Event</u>

Provide a detailed chronology of the January 20, 1986 Unit 3 drywell expansion gap fire occurrence and describe the sequence of events that led to the decision that offsite fire department assistance was not needed.

b. <u>Duration and Intensity of the Fire</u>

Determine the duration, physical extent, and intensity of the fire and include in this assessment the highest metal and concrete temperatures reached during the fire. If no systematic approach was taken to record actual temperatures reached during the fire, determine the highest temperature that the steel and concrete structures may have been exposed to based on published (i.e. Underwriters Laboratories Inc., Factory Mutual Laboratories Inc.) free burning polyurethane foam calorific heat values for a fire of this duration. Provide an estimate of what changes occurred in the material properties of the steel, concrete, electrical and pipe penetrations, drywell penetration wells and other affected equipment or components.

For the normal operating and accident condition, determine the temperature profile through the drywell steel liner with and without polyurethane present in order to show any changes in drywell expansion from the original design. Perform a structural analysis which evaluates the state of stress of the drywell steel liner during the fire and compare this with the yield strengths of the material.

c. Corrosive Species Introduced Into the Drywell Expansion Gap

Determine the type and quantity of corrosives that were introduced into the drywell expansion gap as a result of the fire and its extinguishment. Determine the short and long term effects of these corrosive species on the structural integrity of the drywell steel liner structural and shielding concrete, electrical and pipe penetrations, drywell penetration welds and other affected equipment and components.

d. Effects of Spalling Concrete and Polyurethane Residue Remaining Inside the Drywell Expansion Gap

Determine the effects of polyurethane and fiberglass residue as well as "hard spots" that may have been created by spalling concrete into the drywell expansion gap. Determine the effects of potential "hard spots" on the drywell steel liner under pressure and temperature loads during normal operating and accident conditions and determine the compressive strength these "hard spots" must have to be of concern.

e. Amount of Water Applied to the Drywell Expansion Gap to Extinguish the Fire

Determine any thermal shock that may have occurred to the drywell steel liner and determine the amount of water used to extinguish the drywell expansion gap fire; how much of this water was removed; how much remains unaccounted for and what actions will be taken to remove any remaining moisture in the drywell expansion gap or in the surrounding structural and shielding concrete.

f. Basic Drywell Liner and Structural and Shielding Concrete Design Functions

Determine to what extent (if any) the fire may have otherwise degraded the drywell steel liner's ability to provide a barrier which controls the release of fission products to the secondary containment. Determine to what extent if any, the fire may have otherwise degraded drywell electrical or pipe penetrations and the structural and shielding concrete design functions.

g. Compliance with the Safe Shutdown Requirements of Appendix R to 10 CFR Part 50

Determine the effects of a fire of this nature on safe shutdown capability as prescribed in Section III G.2 of Appendix R to 10 CFR 50. During normal operation, this section requires redundant cables, including non-safety circuits that could adversely affect safe shutdown capability that are located in the same fire area outside of the primary containment, to be separated by a 3-hour fire barrier; be encased in a 1-hour fire barrier with automatic fire detection and suppression installed in the fire area; or be separated by a distance of more than 20 feet with no intervening combustible or fire hazards with automatic fire detection and suppression installed in the fire

area. For normal operation of both Dresden Units 2 and 3, explain how such electrical cables and circuits passing through the drywell expansion gap are in compliance with the requirements of Appendix R so that a fire of this nature will not affect safe shutdown capability during normal operations.

h. Potential Repairs Needed

Determine the need for repairs (if any) to the drywell steel liner, structural and shielding concrete, electrical and pipe penetrations or other affected equipment as a result of the fire. Include in this assessment a time frame for completion and the impact of such repairs on normal reactor operations.

i. Results of Water and Polyurethane Residue Samples

Provide the results of any and all extinguishing water and fire residue samples collected as a result of the fire for NRC review.

j. Corrective Actions Taken to Prevent Reoccurrence

Describe in detail the corrective actions that will be taken to prevent fires involving polyurethane material in the drywell expansion gap, including interim measures currently in place.

k. Provide an assessment of the extent and results of the radiolytic and thermal decomposition of materials in the drywell expansion gap in Unit 2 and an estimate of the effects of such decomposition on fire potential and containment structural integrity.

l. Provide a list of other plant locations where polyurethane or other combustible foam materials are installed in concealed spaces. Identify whether these materials were explicitly addressed as part of our fire hazards analysis.

Items a through l above will be tracked as an open item (50-249/86006-01(DRS)).

m. Emergency Preparedness Concerns

Add the phrase, "plant operating staff", to the Unusual Event Emergency Action Level for Condition No. 18. This is an open item (50-249/86006-02(DRSS)).

4. Open Items

Open items are matters which have been discussed with the licensee, which will be reviewed further by the inspector, and which involve some action on the part of the NRC, the licensee, or both. Open items disclosed during the inspection are discussed in Paragraph 3.

5. Exit Interview

The inspectors met with licensee representatives at the conclusion of the inspection on February 7, 1986, and summarized the scope and findings of the inspection. The licensee achknowledged the statements made by the inspectors. The inspectors also discussed the likely informational content of the inspection report with regard to documents reviewed by the inspectors during the inspection. The licensee did not identify any such documents as proprietary. On February 13, 1986, in a telephone conversation with the licensee, additional concerns regarding compliance with the requirements of Appendix R to 10 CFR Part 50 were discussed with the licensee.

Commonwealth Edison
One First National Plaza Chicago Illinois
Address Reply to Post Office Box 767
Chicago, Illinois 60690

May 6, 1986

Mr. James G. Keppler
Regional Administrator
U.S. Nuclear Regulatory Commission
Region III
799 Roosevelt Road
Glen Ellyn, IL 60137

Subject: Dresden Station Unit 3
Response to Inspection Report
No. 50-249/86-006
NRC Docket No. 50-249

Reference (a): Letter from C. J. Paperiello to Cordell Reed dated February 26, 1986.

Dear Mr. Keppler:

This transmittal is in response to the inspection conducted by your staff on January 28, 29 and February 7 and 13, 1986 of circumstances associated with the January 20, 1986 fire in the Dresden Unit 3 drywell expansion gap. Although no violations of NRC requirements were identified during the inspection, reference (a) requested that we respond to the open items identified in Section 3 of the Inspection Report.

The enclosed report provides an overall evaluation of the fire and its consequences. The Appendix to the report specifically addresses the open items from the Inspection Report with the exception of item 3m, Emergency Preparedness Concerns. We are currently reviewing this item in the General Office and will provide a response at a later date.

Section VII of the report provides a Fire Hazards Analysis of the expansion gap and provides the basis for an exemption to Appendix R Section III.G.3. An exemption request is currently being prepared for submittal to NRR.

If you have any further questions on this matter, please direct them to this office.

Very truly yours,

D. L. Farrar
Director of Nuclear Licensing

lm

Attachment

cc: H. R. Denton - NRR
R. A. Gilbert - NRR
NRC Resident Inspector - Dresden

1660K

May 1, 1986

Subject: NLA Letters NL-86-0290 (R.B. Bevan Report on NRR Investigation of Dresden Unit 3 Drywell Fire), NL-86-0324 (February 25, 1986 Letter from J.A. Zwolinski to D.L. Farrar), and NL-86-0325 (February 26, 1986 Letter from C.J. Paeriello to C. Reed)

Attachment: Evaluation for the effects of the Dresden Unit 3 Polyurethane Fire

Mr. J.R. Wojnarowski:

861655

Attached is our response to the subject documents. This report was planned prior to NRC requests for information and is submitted here as requested by R.B. Bevan at a site visit to Dresden Station on February 6, 1986.

It has subsequently been expanded to address the concern of whether Mark I Containments meet the separation criteria of 10CFR 50, Appendix R as requested in the February 25, 1986 letter from J.A. Zwolinski to D.L. Farrar.

Sufficient bases is presented in this report to justify an exemption to Appendix R. An exemption request will be submitted separately.

In the February 26, 1986 letter from C.J. Paeriello to C. Reed the NRC requested additional information outlined in paragraph 3 of the accompanying inspection report. An appendix to the attached report addresses all questions specifically except 3m., Emergency Preparedness Concerns. This item is being addressed by Dresden Station separately.

D.L. Wilgus

D.L. Wilgus

Approved: J.E. Hausman

J.E. Hausman
Dresden/Quad Cities
Project Engineer

DLW/rr
7563D
cc: F.A. Palmer
T.J. Harkabus
D.J. Scott
E.R. Zebus
D.L. Sanderson

The referenced report, "Evaluation of the Effects of the Dresden Unit 3 Polyurethane Fire," is found in FPPDP Volume 13, Section X.11.

Commonwealth Edison
72 West Adams Street, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690 - 0767

July 17, 1986

Mr. James G. Keppler
Regional Administrator
U.S. Nuclear Regulatory Commission
Region III
799 Roosevelt Road
Glen Ellyn, IL 60137

Subject: Dresden Station Unit 3
Drywell Fire-Correction to
Fire Evaluation Report
NRC Docket No. 50-249

References (a): NRC Inspection Report No. 50-249/86-006.

(b): Letter from D. L. Farrar to J. G. Keppler dated May 6, 1986.

Dear Mr. Keppler:

The reference (a) Inspection Report documented your staff's review of the Dresden Unit 3 drywell fire and requested we respond to concerns identified in the report. Our reference (b) transmittal provided our response in the form of a report documenting our evaluation of the fire. We have recently become aware of a condition at Dresden Unit 3 which conflicts with a statement made in our report.

On page 83 of the reference (b) report, as a response to an NRC question, the following statement was made with regard to the use of polyurethane in other plant locations:

> "Both polyurethane and polyethylene have been used as a filler material at the top of block walls and polyurethane is used to seal penetrations. None of the block walls are considered rated fire barriers and in those walls that use polyurethane as a penetration seal, either the wall is not a rated barrier or, if it is, the polyurethane has been replaced with a fire rated or noncombustible material".

This statement is not completely accurate as the polyurethane has not been replaced in all Appendix R rated barriers. This was discovered recently at Dresden at elevation 589'-0" in the Unit 3 reactor building during ongoing Appendix R walkdowns. As a result of this discovery, we are currently conducting additional walkdowns to identify any other areas where previous walkdowns may not have identified polyurethane mechanical penetration seals in Appendix R rated barriers.

All polyurethane mechanical penetration seals discovered in Appendix R rated barriers will be replaced with appropriately rated seals. These activities will be completed in accordance with our existing schedule for sealing Appendix R penetrations at Dresden (completion by end of the next Unit 2, refuel outage). In the interim, we will implement the same compensatory measures we've committed to for other outstanding Appendix R modifications.

We will also be conducting additional walkdowns at Dresden Unit 2 and Quad Cities Units 1 and 2 to identify any similar applications of polyurethane in Appendix R rated barriers. Any problems identified will be resolved in the same manner as described above.

We have discussed this situation with NRR (R. Gilbert, J. Stang) during a telecon on July 14, 1986. During that call, NRR concurred with our proposed resolution of this issue.

If you have any questions regarding this transmittal, please contact this office.

Very truly yours,

D. L. Farrar
Director of Nuclear Licensing

lm

cc: H. R. Denton - NRR
R. A. Gilbert - NRR
R. B. Bevan - NRR
NRC Resident Inspector - Dresden
NRC Resident Inspector - Quad Cities

1855K

Tab 5

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

DEC 21 1987

Docket No. 50-237
Docket No. 50-249

873633

Commonwealth Edison Company
ATTN: Mr. Cordell Reed
Vice President
Post Office Box 767
Chicago, IL 60690

Gentlemen:

This refers to the routine safety inspection conducted by S. G. Du Pont and P. D. Kaufman of this office on October 23 through December 8, 1987, of activities at Dresden Nuclear Power Station, Units 2 and 3 authorized by Operating License No. DPR-19 and No. DPR-25 and to the discussion of our findings with Mr. E. Eenigenburg and others at the conclusion of the inspection.

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.

No violations with NRC requirements were identified during the course of this inspection.

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,

Mark Ring for
W. L. Forney, Chief
Reactor Projects Branch 1

Enclosure: Inspection Report
No. 50-237/87035(DRP);
No. 50-249/87034(DRP)

See Attached Distribution

DEC 24 1987

Commonwealth Edison Company 2 DEC 21 1907

Distribution

cc w/enclosure:
L. D. Butterfield, Jr.,
Nuclear Licensing Manager
E. D. Eenigenburg, Plant Manager
DCS/RSB (RIDS)
Licensing Fee Management Branch
Resident Inspector, RIII
Richard Hubbard
J. W. McCaffrey, Chief, Public
Utilities Division

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report Nos. 50-237/87035(DRP); 50-249/87034(DRP)

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

Licensee: Commonwealth Edison Company
P. O. Box 767
Chicago, IL 60690

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

Inspection At: Dresden Site, Morris, IL

Inspection Conducted: October 23 through December 8, 1987

Inspectors: S. G. Du Pont
P. D. Kaufman

Approved By: M. A. Ring, Chief Projects Section 1C — 12/22/87 Date

Inspection Summary

Inspection during the period of October 23 through December 8, 1987 (Report Nos. 50-237/87035(DRP); 50-249/87034(DRP)).

Areas Inspected: Routine unannounced safety inspection by the resident inspectors on previous inspection items; operational safety verification; monthly surveillance observation; followup of events; licensee event report followup; management meeting; report review; I.E. Information Notices; maintenance; and commissioners tour.

Results: Of the 10 areas inspected, no violations or deviations of NRC requirements were identified in 9 areas; one violation was identified in the remaining area; however, in accordance with 10 CFR 2, Appendix C, Section V.A., a Notice of Violation was not issued (failure to perform Technical Specification fire barrier surveillance within required time period - Paragraph 6.)

DETAILS

1. Persons Contacted

Commonwealth Edison Company

*E. Eenigenburg, Station Manager
J. Wujciga, Production Superintendent
*C. Schroeder, Services Superintendent
*L. Gerner, Superintendent of Performance Improvement
T. Ciesla, Assistant Superintendent - Planning
D. Van Pelt, Assistant Superintendent - Maintenance
J. Brunner, Assistant Superintendent - Technical Services
J. Kotowski, Assistant Superintendent - Operations
R. Christensen, Unit 1 Operating Engineer
G. Smith, Unit 2 Operating Engineer
*E. Armstrong, Regulatory Assurance Supervisor
W. Pietryga, Unit 3 Operating Engineer
J. Achterberg, Technical Staff Supervisor
R. Geier, Q.C. Supervisor
D. Sharper, Waste Systems Engineer
D. Adam, Radiation Chemistry Supervisor
J. Mayer, Station Security Administrator
D. Morey, Chemistry Supervisor
D. Saccomando, Radiation Protection Supervisor
E. Netzel, Q.A. Superintendent
*C. Turley, Station Q.A.
R. Stols, Q.A. Engineer
*R. Janecek, Senior Participant - Nuclear Safety

The inspectors also talked with and interviewed several other licensee employees, including members of the technical and engineering staffs, reactor and auxiliary operators, shift engineers and foremen, electrical, mechanical and instrument personnel, and contract security personnel.

*Denotes those attending the exit interview conducted on December 8, 1987, and informally at various times throughout the inspection period.

2. Review of Previous Inspection Items (92702)

(Closed) Violation (249/87010-01): Reactor water temperature exceeded 212°F with fuel in the reactor and without primary containment integrity, and low power physics tests were not in progress. The maximum temperature reached was 223°F, which is a violation of Technical Specification 3.7.A.2 LCO. The licensee's immediate corrective actions taken were to reduce reactor water temperature below 212°F and establish primary containment. These actions were completed within 22 minutes from the time of discovery. Plant cooldown procedure DOP 1000-3 has been revised to provide instruction to the operator regarding proper computer point selection and monitoring, primary containment requirements, and reactor water cooling requirements. Quiet hours have been instituted to

provide an atmosphere more conducive to turnovers. The NSO and SCRE involved in this event received certain disciplinary action. In addition, the licensee implemented an Error Free Operation Plan designed to achieve error free startups after refuel outages and subsequent operations.

3. Operational Safety Verification (71710 and 71707)

The inspectors observed control room operations, reviewed applicable logs and conducted discussions with control room operators during the period from October 23 to December 8, 1987. The inspectors verified the operability of selected emergency systems, reviewed tagout records and verified proper return to service of affected components. Tours of Units 2 and 3 reactor buildings and turbine buildings were conducted to observe plant equipment conditions, including potential fire hazards, fluid leaks, and excessive vibrations and to verify that maintenance requests had been initiated for equipment in need of maintenance.

The inspectors, by observation and direct interview, verified that the physical security plan was being implemented in accordance with the station security plan.

The inspectors observed plant housekeeping/cleanliness conditions and verified implementation of radiation protection controls. During the inspection, the inspectors walked down the accessible portions of the systems listed below to verify operability by comparing system lineup with plant drawings, as-built configuration or present valve lineup lists; observing equipment conditions that could degrade performance; and verified that instrumentation was properly valved, functioning, and calibrated.

While touring the Unit 2 and Unit 3 High Pressure Coolant Injection rooms, the inspectors observed the following conditions:

- HPCI exhaust line drain pot level switch cover removed and wire broken off. The licensee had issued work request #69207 on September 25, 1987, to repair the switch, however, no work was inprocess during the inspectors walkdown.

- Local station HPCI motor control valves valve position indicating lights not working for the following HPCI valves:

MO-2-2301-9	HPCI pump discharge valve.
MO-2-2301-14	HPCI main pump recirc to torus valve.
MO-2-2301-10	HPCI pump discharge to CST valve.
MO-2-2301-3	HPCI turbine steam supply valve.
MO-3-2301-3	HPCI turbine steam supply valve.

- HPCI motor control valve stem covers missing on the following limitorque valves:

MO-2-2301-3	Work request #58400 was written on 10/3/86 to replace missing stem cover.
MO-2-2301-10	Stem cover is missing and no work request has been written.
MO-3-2301-10	Stem cover is missing and no work request has been written.
MO-3-2301-15	Stem cover is missing and no work request has been written.

The licensee performs a valve position indicating light walkdown on a monthly basis per DOS 040-4, Revision 2. Unit 2's HPCI valve walkdown, which was completed by the licensee on November 15, 1987, denoted no lamp problems. However, the residents walkdown on December 4, 1987, found the above indicating lights not working. Work requests should be initiated to replace the missing limitorque valve stem covers and repair the indicating lights.

The inspectors reviewed new procedures and changes to procedures that were implemented during the inspection period. The review consisted of a verification for accuracy, correctness, and compliance with regulatory requirements.

The inspectors also witnessed portions of the radioactive waste system controls associated with radwaste shipments and barreling.

These reviews and observations were conducted to verify that facility operations were in conformance with the requirements established under Technical Specifications, 10 CFR, and administrative procedures.

The following systems were inspected:

<u>Unit 2</u>

High Pressure Coolant Injection System
Core Spray System

<u>Unit 3</u>

High Pressure Coolant Injection System
Low Pressure Coolant Injection System

<u>Common</u>

Standby Gas Treatment System

4. Followup of Events (92700)

During the inspection period, the licensee experienced several events, some of which required prompt notification of the NRC pursuant to 10 CFR 50.72. The inspectors pursued the events onsite with licensee and/or other NRC officials. In each case, the inspectors verified that the notification was correct and timely, if appropriate, that the licensee was taking prompt and appropriate actions, that activities were conducted within regulatory requirements and that corrective actions would prevent future recurrence. The specific event was as follows:

On December 4, 1987, the licensee reported to NRC Region III that an employee (Stationman) was charged on December 4, 1987, with "possession with intent to deliver" two ounces of cocaine to an undercover metropolitan area narcotics squad agent. The licensee pulled the individual's site security badge, and performed a search of the individual's personal locker onsite with negative results. On December 7, 1987, the licensee performed a drug dog search within the protected area with negative results.

No violations or deviations were identified in this area.

5. Monthly Maintenance Observation (62703, 71710)

Station maintenance activities of safety related systems and components listed below were observed/reviewed to ascertain that they were conducted in accordance with approved procedures, regulatory guides and industry codes or standards and in conformance with technical specifications.

The following items were considered during this review: the limiting conditions for operation were met while components or systems were removed from service; approvals were obtained prior to initiating the work; activities were accomplished using approved procedures and were inspected as applicable; functional testing and/or calibrations were performed prior to returning components or systems to service; quality control records were maintained; activities were accomplished by qualified personnel; parts and materials used were properly certified; radiological controls were implemented; and, fire prevention controls were implemented. Work requests were reviewed to determine status of outstanding jobs and to assure that priority is assigned to safety related equipment maintenance which may affect system performance.

The following maintenance activities were observed/reviewed:

Unit 2

2C Condensate/Booster pump - inboard seal leaking - repair/replace per work request #D69912.

2B Condensate Transfer pump - mechanical seal leaking - repair/replace per work request #D70309.

Unit 3

3-1601-62 Air operated valve - replace solenoid on December 7, 1987, per work request #70591.

Unit 3

SP-87-10-156, Monthly HPCI System Pump Test for the Inservice Test Program. This special test was performed to take vibration data on the HPCI pump. Impeller replacement on the booster pump is being scheduled for the March, 1988 refueling outage.

No violations or deviations were identified in this area.

6. Monthly Surveillance Observation (61726)

The inspectors observed surveillance testing required by technical specifications for the items listed below and verified that testing was performed in accordance with adequate procedures, that test instrumentation was calibrated, that limiting conditions for operation were met, that removal and restoration of the affected components were accomplished, that test results conformed with technical specifications and procedure requirements and were reviewed by personnel other than the individual directing the test, and that any deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel.

The inspectors witnessed portions of the following test activities:

SP-87-10-156, Monthly HPCI System Pump Test for the Inservice Test Program. This special test was performed to take vibration data on the HPCI pump. Impeller replacement on the booster pump is being scheduled for the March 1988 refueling outage.

No violations or deviations were identified in this area.

7. Licensee Event Reports Followup (93702)

Through direct observations, discussions with licensee personnel, and review of records, the following event reports were reviewed to determine that reportability requirements were fulfilled, immediate corrective action was accomplished, and corrective action to prevent recurrence had been accomplished in accordance with Technical Specifications

Unit 2

(Closed) 87029-00: High Pressure Coolant Injection System Inoperable Due to Steam Leak. While performing surveillance testing of the HPCI system on October 1, 1987, with the unit at 97% rated power, a steam leak in the vicinity of the HPCI turbine shaft seal was observed. After discussions between shift supervision and the Operating Engineer, HPCI was determined to be inoperable at 1450 hours on October 1, 1987. Exact cause of the

steam leak is unknown. A possible cause was a momentary misalignment of the HPCI turbine spring loaded labyrinth shaft seals due to the starting vibrations of the HPCI turbine. The HPCI system was run on October 1, 1987 at 1845 hours and on October 2, 1987 at 1345 hours, with no leakage observed; HPCI was then declared operable.

(Closed) 87030-00: Main Steam Safety Valve Setpoints Found Outside Technical Specification Limits Due to Mishandling and Setpoint Drifts. Two Main Steam Safety Valves (Serial No. BK 6290 and BK 6260) removed during the 1987 Unit 2 outage were tested to determine their as-found setpoint. Valve BK 6290, with a design setpoint of 1260 psig, opened cleanly at 1276 psig, thus exceeding the plus or minus one percent tolerance required by Technical Specification 4.6.E. Valve BK 6260 designed to open at 1260 psig, failed to open twice at 1300 psig. On a third attempt, the valve opened cleanly at 1282 psig, which exceeded the T.S. 4.6.E tolerance. Mishandling of the valve during the transport between the drywell and the test boiler caused contact between the shaft and the internal adjustment guide which increased the valve's setpoint. Valves BK 6260 and BK 6290 were overhauled and setpoints adjusted within the one percent tolerance. To prevent future damage in transport, a protective guard for the stem assembly has been fabricated. The guard will be required during transit per procedural change.

(Closed) 87031-00: Reactor Building Ventilation Isolation Start of SBGT System Due to Irradiated Metal on Fuel Cask. Review of this event is documented in paragraph 4 of Region III Inspection Report 50-237/87026(DRP).

(Closed) 87032-00: Reactor Scram Due to Spurious Main Steam Line Low Pressure Signal Caused by Vibration. Onsite followup of this event was conducted and documented under Followup of Events in Region III Inspection Report No. 50-237/87026 Paragraph 4.r.

(Closed) 870034-00: Nonconservative Core Thermal Power (CTP) Calculation Due to Inadequate Calibration Procedure. Three Rosemount 1151 dp feedwater flow transmitters on Unit 2 and one transmitter on Unit 3 were incorrectly calibrated. The transmitters had not been calibrated to compensate for the effects of static pressure span compressure which resulted in a nonconservative error of 0.44% in DP calculations. It is estimated that Units 2 and 3 exceeded the CTP limit for a total of 46 hours since September 1987 and 100 hours since October 1986, respectively. Upon discovery of the calculation error, the transmitters were recalibrated. Dresden Instrumentation Procedure (DIP) 600-1, Feedwater Control Calibration and Maintenance and Dresden Technical Staff procedure (DTS) 8733, Unit 2/3) Computer Feedwater Flow Calibration, have been revised to include the proper calibration curve for the Rosemount 1151 dp transmitter.

(Closed) 86019-01: Unit 2 Reactor Scram From Main Turbine Trip on High Water Level Due to Failure of Feedwater Regulating Valve and Personnel Error. This supplemental report was issued to provide additional root cause information found after the initial LER was submitted. The licensee discovered that the "A" reactor feedwater discharge check valve failure and a personnel error on the part of an NSO for failure to comply with Dresden Operating Procedure (DOP) 040-4, Control Panel Light Bulb Replacement resulted in a reactor scram.

(Closed) 87001-01: UT Indications Found on Primary System Piping Due to Intergranular Stress Corrosion Cracking. This supplemental report was issued to provide the results of additional UT examinations performed on piping welds as a result of the indications reported in the original LER. No additional indications were found upon completion of all UT testing.

Unit 3

(Closed) 87016-00: Primary Containment Group I Isolation and Reactor Scram Due to Apparent Personnel Error. Review of this event is documented in Region III Inspection Report 50-249/87025, Paragraph 4.

The preceding LERs have been reviewed against the criteria of 10 CFR 2, Appendix C, and the incidents described meet all of the following requirements. Thus no Notice of Violation is being issued for these items.

a. The event was identified by the licensee,
b. The event was an incident that, according to the current enforcement policy, met the criteria for Severity levels IV or V violations,
c. The event was appropriately reported,
d. The event was or will be corrected (including measures to prevent recurrence within a reasonable amount of time), and
e. the event was not a violation that could have been prevented by the licensee's corrective actions for a previous violation.

No violations or deviations were identified in this area.

(Closed) 50-249/87018-00: Fire Stop 18 Month Surveillance Interval Exceeded Due to Procedural Deficiency. On September 20, 1987, a review of upcoming surveillances was being performed when it was found that Dresden Fire Protection Procedure (DFPP) 4175-3, "Shutdown Fire Stop/Break Surveillance", was incorrectly classified as due each refueling outage in the surveillance program. The critical surveillance date for this surveillance was November 1, 1985. This surveillance was completed on April 24, 1986, 5 months and 23 days after the critical date. The critical date was missed due to improper categorizing of the 18 month surveillance as a refueling outage surveillance on the computer. Dresden Technical Specification (TS) 4.12.F.1 requires that these fire barrier (stop/break) penetrations be inspected once every 18 months. Failure to perform this Technical Specification surveillance within the required time period is a violation of TS 4.12.F.1 (237/87035-01;

249/87034-01).. However, since Unit 3 was in cold shutdown throughout this period due to an extended refueling outage, the safety significance of exceeding the surveillance date was minimal. Consequently, this violation meets the tests of 10 CFR 2, Appendix C, and no Notice of Violation will be issued. This item is considered closed.

One violation was identified in this area.

8. I.E. Information Notice Followup (92701)

Each of the following I.E. Information Notices (IEN) was reviewed by the Resident Inspectors to verify (1) that the Information Notice was received by licensee management, (2) that a review for applicability was performed, and (3) that if the Information Notice was applicable to the facility, applicable actions were taken or were scheduled to be taken.

(Closed) IEN 87-23: Loss of Decay Heat Removal During Low Reactor Coolant Level Operation. Dresden Units 2 & 3 do not have RHR systems to remove decay heat, so the licensee reviewed the Shutdown Cooling system with regard to the problem identified in this Information Notice. The Shutdown Cooling system takes suction from the Recirculation system. The Recirculation system takes its suction from the reactor vessel annulus area and this piping is not isolatable. The Shutdown Cooling water is returned to the discharge side of the recirculation pump via the LPCI line. Thus, the reactor water level would have to decrease to a point of uncovering the irradiated fuel before a loss of suction could occur. Dresden has five independent categories of reactor vessel water level instrumentation to monitor reactor water level.

(Closed) IEN 87-42: Diesel Generator Fuse Contacts. The licensee had a similar, but less serious, incident occur during a test of the Unit 2 diesel generator in 1974 following a maintenance outage. Dresden utilizes a similar contact arrangement for the diesel generator potential transformer (PT) fuses and connecting cables. The licensee issued DVR 12-2-74-16 and the ensuing inspection revealed burn marks on the B phase contacts and cables (cables charred), indicative of arcing. Poor mating of the fuse and stationary contacts were identified as the cause of the arcing. In May 1974, the licenses made modifications (M12-2-74-32; M12-2-74-33; M12-3-74-48) to the electrical control cabinet doors which enabled them to be screwed closed, thus, insuring a proper mating of the moveable contact finger connects with the stationary contact.

No violations or deviations were identified in this area.

9. Commissioner's Visit

On October 23, 1987, Commissioner Kenneth C. Rogers, accompanied by the NRC Region III Deputy Regional Administrator, visited the Dresden Nuclear Station. While on site, the Commissioner and Region III Management toured the facility with the licensee's corporate and plant management on a familiarization and plant improvement tour. The Commissioner also held meetings with licensee plant management and supervisory personnel. The Commissioner complimented Commonwealth Edison on Dresden's positive progress and direction.

10. Management Meetings

The President, Executive Vice President and other members of Commonwealth Edison Company met with Mr. Victor Stello, Jr., Mr. James M. Taylor, the Region III Regional Administrator, and other NRC representatives in Headquarters on October 28, 1987, to discuss the company's plans to effect sustained performance improvement at the Dresden Nuclear Power Station.

11. Report Review

During the inspection period, the inspectors reviewed the licensee's Monthly Operating Report for October 1987. The inspectors confirmed that the information provided met the requirements of Technical Specification 6.6.A.3 and Regulatory Guide 1.16.

The licensee announced the following Dresden site management changes effective November 25, 1987:

Quality Assurance (QA) Superintendent, M. Jeisy, will assume the position of INPO Coordinator. The new Q.A. Superintendent is E. Netzel, transferring from the Braidwood Station.

12. Exit Interview (30703)

The inspectors met with licensee representatives (denoted in Paragraph 1) informally throughout the inspection period and at the conclusion of the inspection on December 8, 1987, and summarized the scope and findings of the inspection activities.

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. The licensee acknowledged the findings of the inspection.

Tab 6

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

DEC 14 1987

DEC 21 1987

Docket No. 50-237
Docket No. 50-249

Commonwealth Edison Company
ATTN: Mr. Cordell Reed
Senior Vice President
Post Office Box 767
Chicago, IL 60690

Gentlemen:

This refers to the routine safety inspection conducted by Mr. T. J. Ploski and others of this office on November 16-19, 1987, of activities at the Dresden Nuclear Generating Station, Units 2 and 3, authorized by NRC Operating Licenses No. DPR-19 and No. DPR-25, and to the discussion of our findings with Mr. E. Eenigenberg and others of your staff at the conclusion of the inspection.

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.

No violations of NRC requirements were identified during the course of this inspection.

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

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

Sincerely,

W. G. Snell for

W. D. Shafer, Chief
Emergency Preparedness and
Radiological Protection Branch

Enclosure: Inspection Reports
No. 50-237/87037(DRSS);
No. 50-249/87036(DRSS)

See Attached Distribution

 DEC 14 1987

Distribution

cc w/enclosure:
D. Butterfield, Nuclear
Licensing Manager
J. Eenigenburg, Plant Manager
DCD/DCB (RIDS)
Licensing Fee Management Branch
Resident Inspector, RIII
Richard Hubbard
J. W. McCaffrey, Chief, Public
Utilities Division
D. Matthews, EPB, NRR
W. Weaver, FEMA, RV

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-237/87037(DRSS); 50-249/87036(DRSS)

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

Licensee: Commonwealth Edison Company
P. O. Box 767
Chicago, IL 60690

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

Inspection At: Dresden Site, Morris, Illinois
Production Training Center, Braidwood, Illinois

Inspected Conducted: November 16-19, 1987

Inspectors: T. J. Ploski — 12/10/87 Date

G. M. Christoffer — 12/10/87 Date

J. E. Foster — 12/10/87 Date

M. J. Smith — 12/10/87 Date

Approved By: W. Snell, Chief
Emergency Preparedness Section — 12/10/87 Date

Inspection Summary

Inspection on November 16-19, 1987 (Reports No. 50-237/87037(DRSS); No. 50-249/87036(DRSS))

Areas Inspected: Routine, unannounced inspection of the following areas of the Dresden Station's emergency preparedness program: licensee actions on previously-identified items; emergency plan activations; operational status of the program; emergency detection and classification; protective action decision-making; notification and communications provisions; changes to the program; shift staffing and augmentation; training; and audits. The inspection involved four NRC inspectors.

Results: No violations of NRC requirements were identified during this inspection.

DETAILS

1. Persons Contacted

*E. Eenigenberg, Station Manager
*C. Schroeder, Services Superintendent
*R. Holman, GSEP Coordinator
*R. Jeisy, Quality Assurance Superintendent
*T. Gallaher, Quality Assurance Engineer
*S. Stiles, Training Supervisor
*E. Armstrong, Regulatory Assurance Supervisor
*T. Gilman, Emergency Planning Supervisor
R. Mitzel, Shift Engineer (SE)
R. Sitts, SE
T. Palanyk, Station Control Engineer (SCRE)
R. Speroff, SCRE
J. Bowman, Corporate Emergency Planning Staff
L. DeCarlo, Drill Controller
D. Marco, GSEP Training Instructor
W. Reimers, Training Department Staff
K. Licari, Production Training Center

*Indicates those persons who attended the November 19, 1987 exit meeting.

2. Licensee Action on Previously-Identified Items

(Closed) Item Nos. 237/85013-01 and 249/85012-01: Revise Emergency Action Level (EAL) Condition No. 12 for General Emergency to indicate that this emergency class can also be declared based on environs measurements. Emergency Plan Implementing Procedure (EPIP) 200-T1, Classification of GSEP Conditions, has been revised so that a General Emergency can be classified based on a source term derived from field survey teams' measurements. This item is closed.

(Open) Item No. 249/86002-02: Add the phrase "plant operating staff" to the Unusual Event EAL for Condition No. 18. This item resulted from the investigation of the January 20, 1986 drywell fire incident during a plant outage, which was not a classifiable emergency per the licensee's EALs. A review of the EALs listed in EPIP 200-T1, Revision 6, indicated that no EAL had been revised to satisfy the concern expressed in this Open Item. The Dresden Station's EALs were in the latter states of revision at the time of this inspection. These proposed EALs adequately addressed the concern; however, their submittal for NRC review was tentatively scheduled for sometime during the first quarter of 1988. This item remains open.

(Closed) Item Nos. 237/87028-01 and 249/87027-01: Due to a backlog of filing controlled documents at the Mazon Emergency Operations Facility (EOF), a Severity Level IV violation was issued for not maintaining the facility in an adequate state of readiness. A tour of the Mazon EOF during this inspection revealed that all controlled documents were filed,

and adequate permanent administrative support was available to ensure the maintenance of the operational readiness of the EOF. A controlled document room had been completed the week before this inspection and all controlled documents were filed in this new area. The 1988 audit schedule, issued November 18, 1987, included an item to verify document control at the EOF. This audit line item was part of the licensee's commitment in response to the Notice of Violation. This item is closed.

3. Emergency Plan Activations

NRC and licensee records associated with all emergency plan activations that occurred between December 6, 1986 and October 4, 1987 were reviewed. These records included: Licensee Event Reports (LERs); records generated by NRC Duty Officers; Control Room logs; Nuclear Accident Reporting System (NARS) forms completed by onshift personnel following each emergency declaration; the licensee's Deviation Reports; and evaluations of each emergency plan activation that were performed by the GSEP Coordinator.

During this time period, onshift personnel correctly classified thirteen Unusual Events. Based on the LER review, there were no other classifiable events through October 4, 1987. Initial notifications to State and NRC officials were completed within the regulatory time limits following each emergency declaration.

Based on the above findings, this portion of the licensee's program was acceptable.

4. Operational Status of the Emergency Preparedness Program (82701)

a. Emergency Plan and Implementing Procedures (Also 82204)

The licensee's procedures for the preparation, review and distribution of new and revised EPIPs were basically unchanged from the previous inspection and were adequate. A review of six EPIP revisions indicated that proper procedures were followed to incorporate these revisions into the program. Review of the EPIP Distribution Transmittal Log showed that revisions were distributed to licensee personnel and the NRC within one week after approval. However, EPIPs sent to NRC Region III were not being tracked to ensure receipt of the revisions. When this was mentioned to the licensee, Region III was added to that portion of the EPIP Distribution Transmittal Log that tracks receipt of documents.

Current copies of the emergency plan and implementing procedures were readily available in the Control Room, TSC and EOF.

Based on the above findings this portion of the licensee's program is acceptable.

b. Readiness of Emergency Response Facilities and Supplies (Also 82204)

A tour of Emergency Response Facilities (ERFs) indicated that the Technical Support Center (TSC), Operational Support Center (OSC), and Emergency Operations Facility (EOF) were maintained in an adequate state of operational readiness. Plans, procedures and drawings were filed; communications equipment was operational; and adequate supplies were available.

A controlled document area had been constructed in the EOF to contain plans, procedures and drawings relevant to all of the licensee's nuclear stations. Upgraded equipment was scheduled for installation in the EOF to bring its layout and computer capabilities equal to that of the Zion Station's EOF, which was utilized in the 1987 Federal Field Exercise. Included in this upgrade will be electronic status boards, a PRIME computer system for administrative functions, and an upgraded plant data computer system. A dual purpose transportation facility had been constructed next to the EOF. This facility will house a dedicated "GSEP Van" for offsite survey team use. The structure was also constructed to accommodate overflow EOF and Joint Public Information Center (JPIC) personnel. A limited number of telephone outlets will be available for temporary use by overflow personnel. The re-modeled JPIC was toured and appeared to be about 90 percent complete at the time of this inspection.

A review of 1987 records for emergency equipment and supplies inventories was performed. All required inventories had been completed and adequately documented as required by EPIPs. Inventory records indicated that identified deficiencies had been promptly corrected.

An "Emergency Response Telephone Directory" had been developed for use in the ERFs. This directory contained instructions on use of different communication systems and the business telephone numbers for offsite licensee personnel and offsite support agencies. This computerized directory was scheduled for quarterly review and update by Corporate emergency planning personnel.

An inventory of the supplies in the decontamination and medical facility was requested per relevant EPIPs. The following discrepancies were found:

- EPIP 500-4 stated that there is one portable eye wash device located in the medical and decontamination area. One portable eye wash was in that area; however, it was questionable whether its operability had been checked during the inventory process. When this concern was brought to the attention of the Radiation Chemistry Foreman, he immediately contacted the Operations Department to find out if they had conducted surveillances on this piece of equipment. The Operations Department reported that no monthly operability check had been conducted. They agreed to add this particular portable eye wash to the monthly surveillance schedule.

Several minor differences were noted between the actual contents of the No. 36 first aid kits as described in EPIP 500-1, "Inventory Sheet for First Aid Kits." When told of the discrepancies, the GSEP Coordinator stated that he was in the process of upgrading the No. 36 first aid kits to the Corporate kit standards as stated in CECo General Procedure 826.

Additionally, it was observed that the items listed in the Medical and Decontamination Area Inventory were stored in a disorganized manner in various locations in the room.

Based on the above findings, this portion of the licensee's program was acceptable; however, the following item should be considered for improvement:

- The licensee should arrange supplies kept in the decontamination and medical facility in an organized manner.

c. Organization and Management Control (Also 82204)

The licensee's "Strategic Plan for Excellence in Nuclear Operations, 1988-1992" included the objective of maintaining an effective emergency preparedness program in terms of plans, procedures, personnel, and facilities. With respect to personnel at the Dresden Station, this corporate plan has been translated into a "Basic Expectations for Management Personnel." Supervisors would be held responsible for ensuring that their staffs complete all their emergency preparedness training as scheduled, and that routine activities, such as drills, surveillances, and inventories are adequately done.

The GSEP Coordinator was a Dresden Station employee. The GSEP Coordinator's reporting chain has changed since the last inspection. He formerly reported to the Services Superintendent through the Regulatory Assurance Supervisor. He now reports to that individual through the Rad Chem Supervisor. This change was made in order to be consistent with the licensees' other nuclear stations and to facilitate the coordinator's interface with Rad Chem Department staff, which includes the former GSEP Coordinator.

During this inspection the corporate emergency planning staff received approval to expand its scope of responsibilities with the addition of a position titled GSEP Staff Coordinator, who will report to a corporate Emergency Planning Supervisor based at the Mazon EOF. The GSEP Staff Coordinator's responsibilities will include the coordination of Corporate and Stations' GSEP programs, including training. The coordination of Station and corporate GSEP training efforts and interface with the nearby Production Training Center had been a responsibility of emergency planners based in the licensee's corporate offices in downtown Chicago.

Letters of Agreement with offsite agencies were current. Annual radiological emergency response training for these agencies was conducted, as required by 10 CFR 50.47(b), on September 3, 1987. The agenda included a review of EALs, Emergency Classifications, and information on requesting QA Department assessments of the Station's interface with State and local response agencies. Agenda materials had been revised and upgraded since 1986. A "Nuclear Power Handbook" pamphlet, containing plant systems diagrams and fundamentals of radiation information, plus a booklet on EALs and ERFs were distributed to meeting attendees. Visual aids were also upgraded, and included a film of the plant site and photos of the TSC and EOF. A tour of plant facilities was also offered. This training program improvement was the result of a coordinated effort by Station and corporate personnel.

Based on the above findings, this portion of the licensee's program was acceptable.

d. Emergency Preparedness Training (Also 82206)

The annual training of onsite emergency response personnel had been completed and adequately documented, with the exception of two individuals, whose training was scheduled before December 31. The 1987 training had been accomplished utilizing a combination of EPIPs and training modules relevant to specific positions in the onsite emergency organization. Examinations on training materials were reviewed and found to be adequate.

By memo dated February 3, 1987, the Dresden Station's Training Department was provided with a set of training modules that had been refined from an earlier version by staff at the licensee's Braidwood Station. Site-specific adjustments to these modules had then been made for the Dresden Station. Section 8.2 of the generic GSEP described an "approved GSEP Training Matrix" which delineates the training applicable to specific emergency organization positions. (The matrix of 1987 onsite training requirements had not been formally approved. This was a finding of an October 1987 Quality Assurance (QA) Department Audit No. P-87-IV.) The station's training matrix was formally approved by appropriate Dresden Station management on November 19, 1987. The offsite GSEP Training Matrix was approved on November 20, 1987.

Nineteen of thirty-five training modules were reviewed for inconsistencies with the GSEP and relevant procedures. A discrepancy regarding the Station Director's undelegatable responsibilities is described in Paragraph 6 of this report. The only other problem identified was that another module indicated that a field survey team was to retreat if it encountered a radiation field of at least 100 mR/hour. This guidance was inconsistent with that found in Procedure EG-3, Revision 6, which indicated that a team must immediately inform an Environs Director when encountering a radiation field of at

least 100 mR/hour. When the licensee was informed of these training module errors, both were adequately corrected prior to the exit interview.

Interviews were conducted with five members of the onsite emergency organization. Personnel were adequately knowledgeable of their emergency responsibilities. Additional details regarding the walkthroughs of Control Room personnel are provided in Sections 5 and 6 of this report.

Records review indicated that all required drills had been conducted, critiqued, and adequately documented for the period October 1986 through September 1987. The licensee's evaluation of the September 1987 exercise had also been adequately documented. The final critique report for the November 1987 Medical Drill was not yet available for review.

On November 17th, an inspector observed a semi-annual Inplant Health Physics drill and subsequent critique. A Rad Chem Foreman and two technicians participated in the drill, which was evaluated by two controllers. The response of the inplant team was realistic, with minimal simulation of protective clothing, special dosimetry, radiation survey devices, and communications equipment. An air sample and a number of smear samples were collected. An adequate critique was conducted after the drill. Player feedback was encouraged.

Based on the above findings, this portion of the licensee's program was acceptable.

e. Independent Reviews/Aduits (Also 82210)

Records of the Quality Assurance (QA) Department audits and surveillances since August 1986 were reviewed. All records were readily available and complete. Two audits and seven surveillances were conducted in 1987. Surveillance topics included: drill and exercise evaluations; document control at the Mazon EOF; and responses to two actual emergency plan activations. Audits and surveillances were adequate in scope and depth. The regulatory requirements of 10 CFR 50.54(t) were adequately addressed. The adequacy of interface between the Station and various governmental agencies was also assessed as adequate per Audit AA-87-23. The QA Department adequately tracked corrective action taken on audit and surveillance findings and recommendations. A report of corrective action taken or planned is required within 30 days. The QA Department then conducts a followup audit in 90 days to evaluate the effectiveness of the action taken.

A review of the GSEP Coordinator's informal tracking system for corrective actions to be taken on identified drill and exercise improvement items was conducted. The tracking system was current up to the September 1987 exercise. The corrective actions taken on earlier drill items were adequately documented.

The GSEP Coordinator conducted thorough reviews of all internal documentation associated with all emergency plan activations since the last inspection. The informal review procedure has been upgraded, as the Coordinator also determined whether a declaration was appropriate, rather than only focusing on the timeliness and completeness of the various notifications. The review procedure included provisions for informing Station management and a corporate emergency planning supervisor of any identified problems.

Based on the above findings, this position of the licensee's program was acceptable.

5. Emergency Detection and Classification (88201)

EALs contained in EPIP 200-T1 were consistent with those listed in the current revision of the Dresden Annex to the GSEP. Personnel from the Dresden and Quad Cities Stations and corporate emergency planning staff have been meeting to substantially revise and standardize both Stations' EALs. Although a recent draft of the proposed EALs was available, the licensee did not expect the revised Dresden Station EALs to be submitted for NRC review and approval until the first quarter of 1988.

Two walkthroughs were conducted with Control Room personnel. Each walkthrough involved a Shift Engineer (SE) and a Station Control Room Engineer (SCRE). Both SEs clearly understood that they had the undelegatable responsibility to declare an emergency. Both sets of personnel adequately demonstrated the capability to properly classify abnormal situations in accordance with the Station's EALs. The individuals were adequately familiar with regulatory requirements and procedural guidance for informing State and NRC officials following any emergency declaration.

Based on the above findings, this portion of the licensee's program was acceptable.

6. Protective Action Decisionmaking (82202)

Procedural guidance regarding onsite and offsite protective action decisionmaking was consistent with that found in the current GSEP and Dresden Annex. The locations of onsite assembly areas identified in EPIP 300-3 (Assembly and Evacuation of Personnel) were identical with those shown on an emergency information card made available to personnel granted unescorted access privileges. During observation of an inplant Health Physics drill, the inspector noted that the Unit 2 trackway area was adequately marked as an assembly area. Signs giving directions to this assembly area were readily visible on the 570-foot elevation of the Unit 2 portion of the Reactor Building, on building elevations between that level and the ground level assembly area, and on stairways leading to the Unit 2 trackway.

During the walkthroughs with both sets of SEs and SCREs, it became apparent that there was some uncertainty regarding whether the decision to recommend offsite protective actions and/or the authorization of

emergency worker exposures were undelegatable responsibilities of the Acting Station Director, as stated in the GSEP. The interviewees were assured that both items were undelegatable responsibilities per the GSEP.

A review of relevant procedures and lesson plans uncovered several inconsistencies with the GSEP regarding undelegatable responsibilities. EPIP 100-C1 (Station Director) and EPIP 300-4 (Emergency Personnel Dose Limits) did not clearly indicate that authorization of exposures in excess of 10 CFR Part 20 limits was an undelegatable responsibility. The relevant lesson plan indicated that the declaration of an emergency was the only undelegatable responsibility of the Acting Station Director. These procedural and training program inconsistencies were brought to Station management's attention during the inspection. Both procedures and the relevant lesson plan were revised, and approved per Station procedures, prior to the November 19th exit interview. Onshift personnel would be informed of the procedure changes through the required reading program.

Based on the above findings, this portion of the licensee's program was acceptable.

7. Notifications and Communications (82203)

A review of test documentation for the period April-October 1987 indicated that the licensee has adequately maintained the Prompt Notification (siren) System utilized by offsite officials to alert the public within the 10-mile Emergency Planning Zone (EPZ) of a serious emergency situation at the Dresden Station.

The Station's annual emergency communications test was conducted on February 3, 1987. Documentation was adequate, including indications of prompt corrective actions taken on a few minor problems. Records of periodic communications tests for the period January-November 1987, plus a random testing of equipment during the inspection, indicated that portable communications equipment and fixed equipment in the emergency response facilities had been adequately maintained.

EPIP 500-7 (Nuclear Accident Reporting System (NARS) Test Checklist) was used to document the monthly test of this dedicated system for notifying State and local officials. Completed copies of the checklist also contained handwritten, commercial telephone numbers for the various agencies. However, incorrect commercial numbers had been written on the checklists for the Illinois Department of Nuclear Safety, Will County ESDA, and the Kendall County Emergency Operations Center. Although the procedure required the licensee's caller to test the NARS and to write backup telephone number information on the checklist, it did not require the caller to verify the backup telephone numbers during the NARS test.

Consequently, over a period of time, several incorrect backup telephone numbers had appeared on the completed checklists. Correct backup telephone numbers for the locations in question were contained in appropriate EPIPs and the GSEP Telephone Directory, both of which would be available to emergency response personnel.

A tour of the Unit 2/3 Control Room indicated that a copy of the NRC Duty Officer's Event Notification Worksheet had been placed in the "Red Phone Logbook" for use as a reference when onshift personnel would communicate with the Duty Officer. The licensee indicated that the intent was to improve the quality of communications with the NRC, as onshift personnel could better anticipate the agency's information needs.

Based on the above findings, this portion of the licensee's program was acceptable; however, the following item should be considered for improvement:

- If completing EPIP 500-7 is not intended as a means of verifying backup telephone numbers, then the licensee should delete the procedural requirement to write-in such data on this checklist.

8. Shift Staffing and Augmentation (82205)

The numbers and types of persons required for augmentation of onshift personnel following declaration of a given emergency class were specified in Section 4 of the GSEP, EPIP 300-1, and in a prioritized callout list.

Augmentation provisions met the criteria in Table B-1 of NUREG-0654, Revision 1. Augmentation of onshift personnel is initiated through an Operations Duty Supervisor. The callout lists have been updated on a quarterly basis.

The licensee conducted quarterly drills during 1987 which successfully demonstrated the capability to adequately augment onshift personnel in a timely manner. The quarterly off-hours drills were done in accordance with a commitment in the Dresden Annex to the GSEP. The generic GSEP contained a semi-annual commitment to conduct such drills.

Based on the above findings, this portion of the licensee's program was acceptable.

9. Exit Interview

On November 19, 1987 the inspectors met with those licensee representatives denoted in Paragraph 1 to present their preliminary inspection findings. The licensee indicated that none of the matters discussed were proprietary in nature.

Tab 7

DRESDEN 2 & 3

FIRE PROTECTION PROGRAM DOCUMENTATION PACKAGE

Inspection Report No. 50-237/88010 and 50-249/88012

D. Bliss

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

JAN 3 1989

Docket No. 50-237
Docket No. 50-249

Commonwealth Edison Company
ATTN: Mr. Cordell Reed
Senior Vice President
Post Office Box 767
Chicago, IL 60690

Gentlemen:

This refers to the special safety inspection conducted by Messrs. J. Holmes, R. Hodor, and K. Parkinson of this office on April 18-22, May 11-13, August 15, and December 13, 1988, of activities at Dresden Nuclear Power Station, Units 2 and 3, authorized by NRC Operating Licenses No. DPR-19 and No. DPR-25, and to the discussion of our findings with Mr. E. D. Eenigenburg at the conclusion of the inspection. This inspection was conducted to assess compliance with 10 CFR 50, Appendix R, and to review implementation of certain Fire Protection Program requirements.

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.

During this inspection, certain of your activities appeared to be in violation of NRC requirements, as described in the enclosed Notice. With respect to Item 1 of the Notice, the inspection showed that actions had been taken to correct the identified violation and to prevent recurrence. Our understanding of your corrective actions are described in Paragraph 3.g of the enclosed inspection report. Consequently, no reply to this violation is required and we have no further questions regarding this matter at this time. Regarding the remaining item, which concerns the improper storage of combustible liquids, a written response is required.

Two Level IV's
No respon
30 day respon

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

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

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

Sincerely,

J. Hamlin /for

Hubert J. Miller, Director
Division of Reactor Safety

Enclosures:
1. Notice of Violation
2. Inspection Reports
   No. 50-237/88010(DRS);
   No. 50-249/88012(DRS)

cc w/enclosures:
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NOTICE OF VIOLATION

Commonwealth Edison Company
Dresden Nuclear Station

Docket No. 50-237
Docket No. 50-249

As a result of the inspection conducted during April 18-22, May 11-13, August 15, and December 13, 1988, and in accordance with the "General Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (1988), the following violations were identified:

1. Section III.L of Appendix R to 10 CFR Part 50 requires that alternate shutdown capability provided for a specific fire area shall be capable of maintaining the reactor coolant level above the top of the core. In addition, supporting functions shall be capable of providing the process cooling lubrication, etc., necessary to permit the operation of the equipment used for safe shutdown functions. Further, Section III.L requires that procedures shall be in effect to implement this capability. The licensee was required to establish these procedures by July 19, 1985.

   Contrary to the above, during the inspection conducted on April 18-22, 1988, an NRC inspector identified that no administrative procedures or controls were in effect to insure that required alternative shutdown equipment (i.e., control rod drive pump, service water pump, 4Kv Bus, 480 Bus and 480V MCC) was available for the operating unit when the opposite unit (which houses the alternative shutdown equipment) was in an outage or shutdown and the required alternate shutdown equipment was removed from service for scheduled maintenance or repair.

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

2. 10 CFR 50.48(a) requires that each operating nuclear power plant have a fire protection plan that satisfies Criterion 3 of Appendix A to 10 CFR Part 50. It further requires that the plan shall describe specific features necessary to implement the program such as administrative controls and the means to limit fire damage to structures, systems, or components important to safety so that the capability to safely shutdown the plant is ensured.

   Section B.2 of the licensee's response to the Guidelines of Appendix A to APCSB 9.5-1 as accepted in the 1980 Supplemental Safety Evaluation Report indicates that effective administrative measures will be implemented to prohibit bulk storage of combustible materials inside or adjacent to safety-related buildings or systems during operation or maintenance periods.

   Contrary to the above, during a previous inspection conducted on April 12, 1988, an NRC inspector observed twenty 55-gallon drums of lubricating oil stored in a safety-related area on Elevation 517' - 6" (in the southwest corner) of the Unit 2 Reactor Building. This condition existed from March 31 to April 13, 1988.

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

With respect to Item 1, the inspection showed that action had been taken to correct the identified violation and to prevent recurrence. Consequently, no reply to this violation is required and we have no further questions regarding this matter. With respect to Item 2, pursuant to the provision of 10 CFR 2.201, you are required to submit to this office within 30 days of the date of this Notice a written statement or explanation in reply, including for this violation: (1) corrective action taken and the results achieved; (2) corrective action to be taken to avoid further violations; and (3) the date when full compliance will be achieved. Consideration may be given to extending your response time for good cause shown.

Dated

J J Harrison /for
Hubert J. Miller, Director
Division of Reactor Safety

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-237/88010(DRS); 50-249/88012(DRS)

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

Licensee: Commonwealth Edison Company
Post Office Box 767
Chicago, IL 60690

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

Inspection At: Dresden Site, Morris, Illinois

Inspection Conducted: April 18-22, May 11-13, August 15, and December 13, 1988

Inspectors: J. Holmes
J. Holmes — 12/27/88 Date

J. Holmes for
R. Hodor (BNL) — 12/27/88 Date

J. Holmes for
K. Parkinson (BNL) — 12/27/88 Date

Approved By: R. Gardner, Chief
Plant System Section — 12/28/88 Date

Inspection Summary

Inspection on April 18-22, May 11-13, August 15 and December 13, 1988 (Reports No. 50-237/88010(DRS); 50-249/88012(DRS))
Areas Inspected: Special, announced inspection conducted to assess plant compliance with 10 CFR Part 50, Appendix R and to review implementation of certain Fire Protection Program requirements. The inspection was performed in accordance with NRC Manual Chapter Procedures 30703, 64100, and 64704.
Results: Of the areas inspected, two apparent violations were identified. The licensee has developed a safe shutdown methodology to prevent fuel clad damage or rupture of any primary coolant boundary in the event of a disabling fire in the plant. However, the methodology chosen by the licensee does not incorporate a dedicated safe shutdown panel for a disabling fire requiring the evacuation of the control room, but relies on many manual actions to achieve

safe shutdown conditions. The strength or weakness of this program in achieving its goals in safely shutting down the reactors will be dependent upon good operator training, prudent use of administrative controls and maintaining the present fire protection systems. Weaknesses observed included the following: (1) licensee did not provide administrative controls to insure that the required opposite unit equipment was available for the operating unit when the required opposite unit equipment was down for repair (Paragraph 3.g), (2) administrative controls for combustibles were not effectively utilized in that the licensee permitted the storage of twenty 55 gallon drums of lube oil in a safety-related area where an exemption from the installation of a sprinkler system has been submitted to NRR due to the lack of combustibles in the area (Paragraph 4.b); (3) Unit 1 is no longer operational and does not appear adequately isolated from Units 2 and 3 (Paragraph 2.e.); and (4) in the event of a disabling fire, two hot shorts in multiple conductor cables 33674 and 33934 could cause the spurious operation of the target rock and the electromatic relief valves. While the safe shutdown analysis addresses spurious operation of one valve, the simultaneous spurious opening of the Target Rock Valve and all of the Electromatic Relief Valves has not been analyzed (Paragraph 3.f). Strengths were noted in the application of salient fire protection features between Units 2 and 3 (Paragraph 3.h) and also in the coordination and execution of the fire pump capacity test (Paragraph 2.c).

DETAILS

1. Persons Contacted

Commonwealth Edison Company (CECo)

#*+E. Eenigenburg, Station Manager
#D. Barnett, Quality Assurance
*+B. Barth, Technical Staff Engineer
*W. Betourne, Quality Assurance
#*R. Black, Assistant Fire Marshal
#+J. Brunner, Assistant Superintendent Technical Services
#*R. Christensen, Operations
#*+M. Dillon, Fire Marshal
#+T. Hausheer, Fire Protection Engineer
#*+R. Johnson, Technical Staff Group Leader
#*J. Kotowski, Operations Assistant Superintendent
#+T. Lewis, Regulatory Assurance
*G. Mauropoulos, Boiling Water Reactor Engineer
+E. Netzel, QA Superintendent
#+K. Peterman, Regulatory Assurance Supervisor
*W. Pierce, Engineering Support Service
#D. Roberts, Fire Protection Engineer
*R. Roebert, BWR Engineering
*+C. Schroeder, Services Superintendent
#+J. Silady, Nuclear Licensing Administrator
E. Skowron, Technical Staff Engineer
*+R. Stachniak, Technical Staff Engineer
*J. Wajciga, Production Superintendent
#*+R. Whalen, Tech Staff Mechanical System Group Leader
*J. Williams, Regulatory Assurance

Sargent and Lundy (S&L)

R. Brown, Electrical Engineer
F. Fisher, Electrical Engineer
J. Kelly, Boiling Water Reactor Engineer
C. Ruth, Electrical Engineer

Professional Loss Control (PLC)

*M. Mowrer, Vice President
*C. Ksobiech, Senior Fire Protection Engineer

U.S. Nuclear Regulatory Commission (NRC)

+S. Dupont, Senior Resident Inspector
#D. Jones, Project Inspector
*P. Kaufman, Resident Inspector

*Denotes those attending the April 22, 1988 exit meeting.
+Denotes those attending the October 18, 1988 exit meeting.
#Denotes those attending the December 13, 1988 exit meeting.

2. Licensee Actions on Previous Inspection Findings

a. (Closed) Violation (237/85033-01(DRS); 249/85029-01(DRS)): The licensee failed to consistently and effectively staff the fire protection coordinator position with the result that certain fire protection equipment was not installed, hardware and equipment were not being properly maintained, required training was not completed, and prompt and effective corrective action was not taken for identified deficiencies.

In the licensee's letter dated January 24, 1986, to J. Keppler, NRC, from D. Farrar, CECo, the licensee responded to the violation by indicating that a task force had been assembled to examine the various fire protection duties and tasks that are required to be performed on a company wide basis. The licensee also indicated that, in the interim, a Nuclear Service Technical Fire Protection Engineer from the General Office would assist the station one day per week until the Task Force Report is accepted and implemented. The licensee further indicated that full compliance would be achieved after the Task Force recommendations had been reviewed, evaluated and implemented to the extent deemed necessary.

On July 16, 1986, a followup meeting was held in Region III. In this meeting, the licensee presented several of the recommendations developed by the task force which included providing an Assistant Fire Marshal to the Dresden site and the formation of a fully staffed Corporate Fire Protection Group by late 1987. The licensee indicated that an Assistant Fire Marshal at Dresden was hired as a result of the Task Force recommendation. However, the Task Force recommendations had not been fully implemented and were being reviewed by upper management.

The inspector subsequently requested the licensee to provide a completion date as to when the Fire Protection Task Force recommendations would be implemented. In a letter dated June 10, 1988, from C. Reed, CECo, to A. B. Davis, NRC, the licensee stated that during March 1988 Executive Management had reviewed previous fire protection program assessments and the status of the Fire Protection Task Force Report. The review concluded that increasing the size of the Corporate Fire Protection Group was a desirable enhancement however Executive Management concluded that at that time the group did not need to be as large as recommended by the Task Force Report. The June 10, 1988 letter states, "In summary, Commonwealth Edison believes that the fire protection program deficiencies at Dresden have been corrected and applicable corporate recommendations have been implemented. Further implementation of the corporate Task Force recommendations will be driven by the desire to achieve excellence in fire protection for all our plants including Dresden."

Based on the licensee's actions of assembling a Fire Protection Task Force to examine various fire protection duties on a company wide basis, hiring an Assistant Fire Marshal and implementing applicable Task Force recommendations, this item is considered closed.

b. (Closed) Unresolved Item (237/85033-02(DRS); 249/85029-02(DRS)): The qualifications of the Station Fire Marshal did not appear to be commensurate with the list of responsibilities assigned to that position. The lengthy list of responsibilities constituted a work load that may not have been achievable by a single individual, regardless of the individual's qualifications and experience.

In the January 24, 1986 letter, the licensee indicated that a Task Force had been assembled to examine the various fire protection duties and tasks that are required to be performed on a company wide basis. The Task Force duties included review of the primary responsibilities of the Fire Marshal position. The Task Force recommended a proposed organizational structure for effectively performing fire protection duties in the company. The Task Force indicated that given the numerous duties and responsibilities at the station level, all nuclear stations needed to provide a full-time Assistant Fire Marshal and fire brigade instructor, in addition to the Fire Marshal. Consequently, in a June 10, 1988 letter from C. Reed, CECo to A. B. Davis, NRC, the licensee stated that Commonwealth Edison believed that the fire protection program deficiencies at Dresden had been corrected and applicable corporate recommendations had been implemented.

Based on the licensee's actions of hiring an Assistant Fire Marshal, providing at least one qualified fire brigade instructor to assist the Fire Marshal, and the June 10, 1988 response, this item is considered closed.

c. (Open) Unresolved Item (237/85033-03(DRS); 249/85029-03(DRS)): Several of the licensee's fire protection Technical Specification surveillance procedures did not contain appropriate test requirements and failed to incorporate quality affecting parameters as delineated in NFPA standards.

In the letter dated January 24, 1986 to J. Keppler, NRC, from D. Farrar, CECo, the licensee responded to the items identified by the inspector. The licensee indicated that as a result of an NFPA code review the surveillance procedures would be revised.

During this inspection, the inspector reviewed the revised surveillance procedures that were previously identified as deficient. During this review, the inspector identified the following concerns:

(1) <u>Diesel Fire Pump Testing</u>

(a) The inspector reviewed the updated licensee's diesel fire pump annual capacity check and weekly operability surveillance procedures to verify the automatic operation of the diesel fire pumps. The inspector noted that the annual capacity check procedure did not verify automatic operation of the fire pump. The licensee contended that the fire pump is automatically started at least once a month.

The weekly surveillance procedures direct the testing personnel to automatically start the fire pump by opening the test petcock which is on the side of the fire pump controller. In the 15th edition of the Fire Protection Handbook, Section 16, Chapter 6, Paragraph 5, titled "Annual Pump Test," it indicates that when testing the pumps it is not sufficient to initiate a pressure drop by the test cock on the controller to simulate automatic operation. On June 2, 1988, the inspector informed the licensee that at least once a year, preferably during the annual fire pump test, the automatic mode of the controller should be tested by opening a two inch drain valve (on a fire protection water system riser) or hydrant as inferred from the 15th Edition of the Fire Protection Handbook. The licensee agreed to incorporate into the annual pump test procedure a simulated pressure drop by opening a two inch drain on a fire protection system riser or opening a fire hydrant.

(b) In the Unit 2/3 Diesel Fire Pump Check Surveillance Procedure, it indicates in Section C titled, "Prerequisites" that "If a vibration analysis machine is to be used, the Fire Marshal should contact the cognizant Technical Staff Engineer."

The inspector discussed with the licensee the establishment of vibration analysis baseline data for the diesel fire pumps and conducting the fire pump vibrational analysis test in conjunction with the annual fire pump test. The licensee indicated that the vibrational analysis will be performed as part of the annual fire pump test.

(c) In the NFPA 20 Formal Interpretations, No. 83-2, it indicates that the results of the annual fire pump test should be compared to the manufacture's certified shop test characteristic curve and field acceptance characteristic curve to determine the pump's ability to continue to attain satisfactory performance at peak loads.

During the previous inspection, it was identified by the NRC inspector that the original manufacturer's shop test curve or field acceptance test were not available to the licensee's staff. Since the previous inspection, the licensee has developed fire pump curves from the manufacturer data plates on the fire pump. The licensee has incorporated the developed fire pump curves into their procedures as part of upgrading the Fire Pump Capacity Check Procedure. During this inspection, at the request of the inspector, the licensee performed a capacity check for the Unit 2/3 Diesel Fire Pump. The licensee performed excellently in the coordination and execution of the test No discrepancies were noted from the test results. At the request of the inspector, the licensee agreed to update the procedure to include certain pump parameters such as water jacket temperature and oil pressure.

(2) Testing of Water Suppression Systems

Section 4.12.B.1(e) of Technical Specifications requires that fire suppression water systems be demonstrated operable by performing a system functional test which includes simulated automatic actuation of the systems throughout their operating sequence. The licensee's commitment in Section 3.5.E.3 of the Fire Hazard Analysis (FHA) Report requires that automatic sprinkler systems conform to NFPA Standard No. 13.

The licensee's Surveillance Procedure No. SP 84-6-39 failed to incorporate appropriate test requirements to demonstrate that the sprinkler system is operable in accordance with NFPA 13 in that the procedure did not require flow from the two inch drain valve of wet or dry pipe sprinkler systems.

The licensee indicated to the inspector that the two inch drain test is not conducted because the fire protection water (river water) destroys the radwaste demineralizer beds. The licensee contends that the two inch drain test does not need to be conducted because the fire protection control valves are provided with tampers or locks that ensure an adequate water supply will be available. In addition, the licensee contends that water is available to the sprinkler system because the inspector test is conducted.

The inspector discussed several methods of conducting the two inch drain test that would provide assurance that the system is operable and minimize impact to the demineralizer beds. The inspector informed the licensee that the two inch drain test should be performed and that any deviation regarding the two inch drain test should be adequately justified and documented in the NFPA Code Review Section of the Fire Hazard Analysis.

The licensee indicated to the inspector that resolution to this issue will be provided tentatively by January 1, 1989. Therefore, the unresolved item will remain open pending review and acceptance of the licensee's resolution to this issue.

d. (Closed) Violation (237/85033-04(DRS); 249/85029-04(DRS)): An early warning automatic fire detection system was not installed in the Refueling Floor Area as required by provisions of Amendment No. 36 to Provisional Operating License No. DPR-19 and Amendment No. 33 to Facility Operating License No. DPR-25.

The licensee has requested an exemption from the requirements of providing a fire detection system on the Refueling Floor which is currently being reviewed by NRR. Based on the exemption request, this item is closed.

e. (Closed) Open Item (237/85033-05(DRS); 249/85029-05(DRS)): The licensee agreed to update their response to the NRC and describe the administrative controls and the actions that will be necessary to isolate Unit 1 from Unit 2 and Unit 3 since Unit 1 is no longer operational, but shares common areas with Units 2 and 3. The inspectors also requested that the licensee describe those administrative controls and actions that will be necessary to separate common areas.

In a letter dated January 24, 1986, to J. Keppler, NRC, from D. Farrar, CECo (Responding to the Open Item), the licensee indicated that a stricter transient combustible control procedure would be developed. In addition, a cognizant foreman was designated to assist the Fire Marshal in timely correction of housekeeping deficiencies. The licensee indicated that a detailed memorandum discussing the proper handling of fire barriers had been discussed with all personnel at the station as part of the weekly "tailgate" staff meeting. Also, Procedure No. DFPP 4175-1, Fire Barrier Integrity and Maintenance, has been revised to further clarify the proper handling and maintenance of fire barriers (including fire doors, fire dampers, fire walls, penetration seals) for mechanical and electrical components.

Based on the licensee's updated response, this Open Item is considered closed, although other specific concerns are being raised as described below.

In the letter dated January 24, 1986, to J. Keppler, NRC, from D. Farrar, CECo, the licensee indicated that the separation of Unit 1 was being covered by the Appendix R review program and that this information was being added to the updated FHA for Units 2 and 3. In Section 4.15.9 of the updated FHA the analysis describes that the only portion of the Unit 1 structure which contacts the Unit 2/3 structures is the west wall of the Unit 1 Turbine Building. The FHA also indicates that the wall separating the Unit 1 Turbine Building from Auxiliary Electric Equipment Room (AEER) (Fire Zone 6.2)

is a minimum 3 foot 3 inch reinforced concrete three hour fire barrier. The remaining wall west of the Unit 1 Turbine Building has metal siding on unprotected structural steel with openings (non-fire rated doors) that expose Unit 2 Safety Related Areas. The Unit 2 side of this portion of the west wall is identified as Five Zone 8.2.5.A.

During this inspection, the inspectors noted large amounts of RAD worker's clothing and flammable/combustible liquids stored in Unit 1 in an area where if a fire occurred the Unit 2 Fire Zone 8.2.5.A (Safety Related Area) may have been exposed since there is no fire rated barrier between the two areas on Elevation 517'-6". The licensee indicated to the inspector that in the event of a fire from Unit 1 affecting the Unit 2 side (Fire Zone 8.2.5.A) one Safe Shutdown Path would still be available.

The licensee's response would allow a fire to migrate from Unit 1 to Unit 2 AEER and does not appear to be consistent with Section F.18 of the licensee's FHA which indicates that storage areas should be located such that a fire or effects of a fire including smoke will not adversely affect any safety-related systems or equipment.

On December 13, 1988, during a site visit, the inspector was informed by the licensee that the present abandoned equipment in Unit 1 restricts large amounts of combustible storage. The licensee agreed to limit the combustible loading in the area to a low fire load (as defined by plant combustible load procedures) to at least 20 feet from Unit 1 control room walls and the metal wall between Unit 1 and 2. The inspector also observed two non-rated metal doors between Unit 1 and Unit 2 that were maintained in the open position. Unit 1 is currently being Decommissioned and it is expected that combustibles will be stored in Unit 1 and that cutting and welding operations will be performed. It is the inspector's concern that the Unit 2/3 Control Room and Unit 2 Safety Related areas may be exposed to Unit 1 fire since they are not separated by three hour fire walls or other recognized fire protection methods of protecting safety-related areas from adjacent exposures.

This is considered an unresolved item (237/88010-01(DRS); 249/88012-01(DRS)) pending resolution from NRR. The licensee indicated that a three hour fire wall is tentatively scheduled to be installed between Unit 1 and the Unit 2/3 Control Room by December 1989.

3. Assessment of Appendix R Compliance

On a sample basis, the inspectors examined measures that the licensee implemented to assure safe shutdown capability and compliance with 10 CFR Part 50, Appendix R. The inspection consisted of an assessment of the licensee's implementation of Appendix R requirements for physical plant conditions, required operator actions, systems, and components, operator training, supplemental procedures, and methodology employed to mitigate resultant adverse equipment operability due to plant exposure to fires. The results of the inspector's review are as follows:

a. Systems Required for Safe Shutdown

The Appendix R goals required to achieve post-fire safe shutdown are:

- Reactivity control capable of achieving and maintaining cold shutdown reactivity conditions (reactor coolant temperature less than or equal to 200°F).
- Reactor coolant makeup capable of maintaining water level above the top of the core at all times during shutdown operation.
- Reactor pressure control and decay heat removal.
- Process monitoring capable of providing direct readings to perform and control the above functions.
- Supporting functions capable of providing process cooling, lubrication, etc., necessary to permit operation of the equipment used for safe shutdown functions.

In accomplishing the goals outlined above, the equipment and systems used to achieve and maintain hot shutdown conditions should be free of fire damage and capable of maintaining such conditions for 72 hours, using offsite or onsite emergency power. The equipment and systems used to achieve and maintain cold shutdown conditions should be either free of fire damage or the damage to these systems should be limited such that repairs can be made and cold shutdown conditions achieved within 72 hours, using offsite or onsite emergency power.

During the post-fire shutdown, the reactor coolant system process variables shall be maintained within those predicted for a loss of normal ac power, and the fission product integrity shall not be affected; i.e., there shall be no fuel clad damage, rupture of the containment boundary.

(1) Reactivity Control

The licensee takes credit for a reactor trip even for a postulated fire that requires evacuation of the control room. Upon loss of power, or in case of fire damage to the logic circuitry, the system is designed to fail safe (rods fully inserted).

(2) Reactor Coolant Makeup

The isolation condenser method of hot shutdown utilizes the Control Rod Drive (CRD) Hydraulic System to provide makeup to the reactor vessel. One of the two CRD pumps per unit provides all of the reactor makeup required due to leakage and shrinkage during cooldown. The CRD pumps take suction from the condensate storage tank and the condenser hotwell. The CRD pump's discharge

pressure can be monitored locally on mechanical indicators P12(3)-302-73A and P12(3)-302-73B. Local control pushbutton stations have been installed for the CRD pumps. The CRD water headers for the two units are connected with a crosstie line which is normally isolated by manual valves. The valves are located on the mezzanine level of the Turbine Building in the area with accessibility to either set of pumps. Therefore, a fire in one unit will not prevent the other unit's pump from supplying makeup water to the affected unit. The CRD pumps can be cooled by the Service Water system if normal cooling from the Turbine Building Closed Cooling Water (TBCCW) system is lost.

For those fires where the isolation condenser method of shutdown is unavailable, the High Pressure Coolant Injection (HPCI) system is used. The HPCI system consists of a steam turbine driven pump that can take suction from either the suppression pool or the condensate storage tank and pump water to the reactor vessel. The steam that drives the turbine comes from the reactor and is exhausted to the suppression pool. The HPCI system automatically initiates on low-low water level signal (-59 inches) or can be manually initiated from the control room.

The HPCI pump injects water from the condensate storage tank to the reactor vessel. The HPCI system pumps makeup water to the reactor at a rate of 5,600 gpm. The operator can manually operate the flow controller in the control room.

Condensate storage tank level is normally monitored in the control room using level indicators L12/3-3341-3 and L12/3-3341-4. Level can be monitored on mechanical indicators L12/3341-77A and L12/3341-77B located in the Turbine Building in the southeast corner of the Unit 2 reactor feed pump room. If long-term operation of the HPCI system depletes the condensate storage supply, the operator will align the HPCI suction with the suppression pool by opening Valves M02(3)2301-35 and M02(3)2301-36. The HPCI suction is automatically shifted to the suppression pool when the condensate storage tank contains less than 10,000 gallons. HPCI pump discharge pressure can be monitored in the control room on pressure indicator P12(3)2340-2 and locally on mechanical indicator P12(3)-2357.

(3) Reactor Pressure Control and Decay Heat Removal

Initial pressure control and decay heat removal for the reactor is supplied by the electromatic relief valves. However, the target rock valve (mechanical mode) and mechanical safety valves on the steamlines will provide these functions if operation for the relief valves has been affected by a fire. Long term (up to 72 hours) reactor pressure control and decay

heat removal is provided by the isolation condenser system in that the system is sized to handle the total decay heat load five minutes after scram. The isolation condenser consists of two tube bundles in a large water-filled shell. The reactor steam flows through the tubes, is condensed, and returns to the reactor vessel. The water in the shell is boiled off and vented to the atmosphere. The vent line to the main steamline is isolated upon initiation of the isolation condenser system.

If a fire has affected automatic operation of the accessible isolation condenser valves (MO2(3)-1301-2 and MO2(3)-1301-3; Valve MO2(3)-1301-2 is normally open), the operators can remove power from the appropriate motor control centers so that the valves may then be opened by use of handwheels. Normally open valves MO2(3)-1301-1 and MO2(3)-1301-4 are located in the drywell and are therefore not accessible for manual operation. In the event a fire causes these valves to spuriously close, an alternate 480V power feed to each of these valves is provided along with a local control station. In addition, isolation switches have been installed for the normal control and power cables. If the valves spuriously close, the alternate feed is energized and the valves opened. The operator can then deenergize the valves in the open position. Valve MO2(3)-1301-3 is manually throttled to control the cooldown.

Initial makeup to the condenser will be supplied from the condensate storage tanks via the condensate transfer pump. With no makeup, the water stored above the isolation condenser tubes is depleted in 20 minutes after initiation of the isolation condenser system. The isolation condenser level is normally monitored in the control room on level indicator L12(3)-1340-2. The operator can locally monitor the level in the isolation condenser on an existing sight glass by opening two manual valves. Any of the four condensate transfer pumps (two per unit) can supply makeup water to either unit's isolation condenser through the normally open tie line. Therefore, a fire in one unit will not prevent the other unit's pump from supplying makeup water.

When the HPCI shutdown method is used, reactor pressure control and decay heat removal are accomplished by the HPCI turbine (driven by reactor vessel steam) in conjunction with electromatic relief Valves 2(3)-0302-3B through 2(3)-0203-3E. The HPCI turbine steam supply line, the target rock valve, and the electromatic relief valves discharge to the suppression pool. Continued operation of the HPCI system results in heatup of the suppression pool water. One division of Low Pressure Coolant Injection (LPCI)/Component Cooling Service Water (CCSW) is sufficient to remove decay heat from the suppression pool. The operator manually places the LPCI/CCSW system into operation in the torus cooling mode from the control room, thus maintaining

the water temperature within acceptable limits. The operator can also throttle flow as appropriate to obtain the desired cooling. Each LPCI pump is capable of providing a flow of 5,000 gpm. Each CCWS pump is capable of providing a flow of 3,500 gpm.

(4) Process Monitoring

The operator requires a means to ascertain the values of various plant parameters in order to perform required system transitions and essential operator actions. Various process monitoring functions are available to adequately support reactivity control, reactor coolant makeup, pressure control, and decay heat removal as follows:

- Reactor Vessel Level
- Reactor Vessel Pressure
- Suppression Pool Level
- Suppression Pool Temperature
- Condensate Storage Tank Level
- Isolation Condenser Level

Additionally, discharge pressure indication is provided for the CRD pumps, condensate transfer pumps and service water pumps.

Support Equipment

The following equipment is available for post-fire shutdown:

- Emergency Diesel Generators
- 4160V ac
- 480V ac
- 125V dc
- 120V ac
- Communication System
- Emergency Service Water System (ESW)
- Reactor Building Closed Cooling Water System (RBCCW)
- Turbine Building Closed Cooling Water System (TBCCW)
- Containment Cooling Water System (CCSW)
- Service Water System (SWS)
- Fire Water System (FWS)

(5) Cold Shutdown

Two systems are identified at Dresden Station to bring the plant to cold shutdown (reactor coolant equal to or less than 212°F). The preferred shutdown cooling path is the shutdown cooling path which utilizes the Shutdown Cooling System (SDCS). For those fires where the SDCS is not available, LPCI/CCSW is used to achieve and maintain cold shutdown.

The SDCS pumps take suction from the reactor recirculation loops through motor-operated valves 1001-1A and 1001-1B. These valves are inside containment. They are powered from 480VAC MCC 28-1 (38-1) which can be supplied from the emergency diesel generators. They are closed until initiation requirements (reactor coolant system temperature less than 350°F) are met and operator action is taken.

The two inlet lines join in one header outside of containment. This header feeds three separate loops. Each loop has a DC powered motor operated pump inlet isolation valve (1001-2A, 1001-2B, or 1001-2C), a centrifugal pump rated at 6,750 gpm at "full operation," a heat exchanger, and a DC powered motor operated pump outlet isolation valve (1001-4A, 1001-4B, or 1001-4C). Downstream of the pump outlet isolation valves, and still outside containment, the three branches again feed a common header. This common header divides into two return lines, each containing an AC powered motor operated isolation valve (1001-5A and 1001-5B). Each return line penetrates the containment and rejoins the reactor coolant system through connections into one division of the LPCI system. Each LPCI division connects to one of the reactor recirculation loops. Although the capability exists to permit flow from and to both recirculation loops, normally only one loop is selected for such service. Either Recirculation Loop Valve(s) 0202-5A(B) and 0202-7A(B) or 0202-4A(B) must be closed to prevent back flow through the reactor recirculating pump.

The heat exchangers of the SDCS are cooled by water from the RBCCW system, with the heat exchangers of the RBCCW system in turn cooled by the SW system. If the SDCS is not available, the LPCI system can be used to inject cooling water into the core once the injection initiation limits (350 psig) are met. The system is a low pressure, high volume system capable of providing substantial volumes of cooling water to the core. The pump is powered from "emergency" buses, and all motor operated valves are powered from "emergency" MCCs and are also outside containment, accessible for manual operation if needed.

The reactor vessel is allowed to fill using LPCI, overflowing hot water to the pressure suppression chamber (torus) through the relief valves. The continuous cycle of water through the core, through the relief valves to the torus and back again after cooling via the containment cooling heat exchangers, would only be limited by the design of the relief valves themselves. These valves incorporate a spring which must be overridden by system pressure to open the valve. The valve will reseat at approximately 50 psig and will be held shut until the core heats up again and raises pressure, or until the pressure is increased to 150 psig by the LPCI pumps (design head 114 psig at 0 psig reactor pressure to 245 psig at 200 psig reactor pressure). Suppression pool water is pumped through containment cooling heat exchangers and then injected into the reactor vessel.

b. Alternate Shutdown

The licensee has chosen five different Appendix R shutdown paths per unit. Two of the paths per unit have been designated as alternative shutdown paths as described below:

- Path A1 utilizes the Unit 2 pumps and power train by mechanical crossties to shutdown Unit 3 for a fire in Fire Area RB3-II and Fire Area TB-III.
- Path A2 is used to shutdown Unit 2 for a fire in Fire Area TB-V or Fire Area TB-II.
- Path B1 utilizes Unit 3 pumps and power trains via mechanical crossties to shutdown Unit 2 for a fire in Fire Area RB2-II and Fire Area TB-1.
- Path B2 is utilized to shutdown Unit 3 for a fire in the Fire Area TB-V or Fire Area TB-II.

For a fire in the control room or auxiliary electric equipment room requiring control room evacuation, the licensee has developed Procedure EPIP 200-20 for post fire safe shutdown.

c. Procedures for Alternate Safe Shutdown

The licensee has developed Procedure No. EPIP 200-20, Revision 4, dated April 1988, to be used in the event of a fire in the Control Room or the AEER which requires evacuation of the Control Room. A staff of 13 licensee personnel is used to implement the procedure which provides for achieving stable hot shutdown for both Units 2 and 3. A two-column format is used with one column assigning responsibility, and the other column listing the actions required. The procedures include Attachments 1 through 9. Each attachment summarizes the actions for an individual operator. After stable hot shutdown conditions have been achieved, Procedure DSSP 200-5 is entered to bring the units to cold shutdown.

Once the decision to evacuate the control room is made, the reactors are tripped from the control room driving the control rods in for hot shutdown reactivity control. Several other immediate actions are attempted in the control room prior to evacuation; however, if unsuccessful, they are covered by procedure from outside the control room after the evacuation.

The scope of the team review was to ascertain that post-fire safe shutdown using the steps in the procedure could be attained in a safe and orderly manner, while achieving the functional goals of Appendix R. No unacceptable items were found by the team review of the procedure.

A walkdown of Procedure No. EPIP 200-20, Revision 4, April 1988, "Control Room Evacuation/Safe Shutdown," was conducted on April 20, 1988, at 1300 hours. The purpose of the walkdown was to determine by simulation that alternate safe shutdown could be implemented in a safe and orderly manner for a fire in the Control Room or AEER. Four inspectors accompanied the operators during the walkdowns.

The following conditions were specified for the simulated shutdown.

- Reactor at 100% power with systems lined up in normal full power configuration.

- Credit for one manual action prior to evacuating the Control Room.

- Loss of offsite power.

- Manual start of emergency diesel generator.

The team paid particular attention to the feasibility of each manual action, ease of access, operator familiarity with procedural steps and equipment, communications, emergency lighting, and the direction of the shutdown by the shift engineer. The walkdown was halted when the licensee had adequately demonstrated the capability to achieve simulated stable hot shutdown conditions.

No unacceptable items were identified by the team during the walkdown. However, in subsequent discussion with the shift engineer the licensee was informed that a visual aid, showing on a single page the flow of actions for each of the nine Individual Operator Attachments, would facilitate the shutdown training provided by the shift engineer. The licensee agreed to implement this recommendation.

- <u>Hot Shutdown Repairs and Manual Actions</u>

  The licensee has identified in Section 7.3 of the Dresden Fire Protection Documentation Package entitled, "Procedures Relevant to Hot Shutdown," hot shutdown repairs and manual actions necessary to achieve hot shutdown. NRR has reviewed the identified hot shutdown repairs and manual actions in the July 17, 1987 SER. Approval was granted contingent upon verification by the inspection team. The team review conducted during the April 18-22, 1988 audit focused on verifying that the necessary actions can be completed within the specified times for assuring safe shutdown.

  Based on a detailed review of the Dresden safe shutdown procedures, including a walkdown of the EPIP 200-20 procedure for shutdown outside the Control Room, the inspection team determined (taking into account the licensee's available manpower for post-fire safe shutdown - 11 personnel exclusive of fire brigade) that post-fire safe shutdown can be

accomplished. This included the initiation of makeup to isolation condenser shell within 20 minutes, and closure of a spuriously opened relief valve within 10 minutes.

d. Operator Training on Safe Shutdown Procedures

In addition to observing the operator's performance during the walkdown, training personnel were interviewed and lesson plans reviewed concerning operator training on Appendix R post-fire safe shutdown procedures and equipment. Training records for operating shift personnel were also reviewed. The areas reviewed were found to be satisfactory.

e. Protection for Associated Circuits

The licensee's associated circuits analysis was provided in Dresden Station Units 2 and 3 Fire Protection Program Documentation Package, Volume 3, Book 1, Section 3.3, Associated Circuits.

The following associated circuits were evaluated:

- Common Bus Concern

  The common bus associated circuit concern is found in circuits, either safety-related or non safety-related, where there is a common power source with shutdown equipment and the power source is not electrically protected from the circuit of concern.

- Spurious Signals

  The spurious signals concern is made up of two items:

  - The false motor, control, and instrument readings such as those which occurred at the 1975 Browns Ferry fire. These could be caused by fire initiated grounds, shorts, or open circuits.

  - Spurious operation of safety-related or non safety-related components that would adversely affect safe shutdown capability (e.g., RHR/RCS isolation valves).

- Common Enclosure

  The common enclosure associated circuit concern is found when redundant circuits are routed together in a raceway or enclosure and they are not provided with adequate electrical isolation protection, or fire can destroy both circuits due to inadequate fire protection methods.

  The inspection results were as follows:

(1) Common Bus Concern

The common bus concern consists of two items:

- Circuit Coordination
- High Impedance Fault Analysis

(a) Circuit Coordination

Breaker Coordination is audited by reviewing the time current curves developed during the licensee's bus coordination study. Licensee representatives stated that the original plant design provided circuit coordination. However, documentation demonstrating coordination of electrical devices was not provided to the electrical inspector. Additionally, licensee representatives stated that in the 480V distribution systems circuit coordination does not exist for some circuits. The licensee's analysis identifies the lack of coordination between the 480V Switchgear Buses 18, 19, 28, and 29 main feeds to MCCs and the motor control branch circuits.

Based on the existing lack of coordination for 480V MCCs, the lack of readily available records, the lack of coordination curves demonstrating coordination, and the requirement to provide protection in the case of high impedance faults and spurious operations, the licensee has provided circuit coordination by manual operations. The manual operations specified in procedures include: circuit breaker, disconnect, and switch operations and fuse removal.

The following circuits were randomly selected for review to verify that circuit coordination was provided procedurally:

CIRCUIT	COMMENT
4kV Bus 23	Coordinated by procedures
4kV Bus 24	Coordinated by procedures
480V Bus 38	Coordinated by procedures
480V Bus 39	Coordinated by procedures
480V MCC 28-2	Coordinated by procedures
125V DC Bus 3A	Coordinated by procedures
125V DC Panel No. 2	Coordinated by procedures

Manual credit for breaker coordination was found to be satisfactory.

Control of fuse replacement is required to ensure maintenance of coordination for circuits protected by fuses. The licensee does not have an established program or procedure for controlling fuse replacement. By memo dated April 20, 1988, the licensee promulgated the following policy on replacing blown fuses:

- Compare the new fuse to the old fuse to verify that they are "like for like." This comparison should include manufacturer, physical size, shape, voltage rating, current rating, and fuse type (quick-acting, slow-blow, etc.).

- If illegible or missing markings on the old fuse do not permit a complete verification of voltage and current ratings or fuse type, the Shift Supervisor will obtain verification of such data from wiring diagrams and/or vendor manuals or by consultation with the Technical Staff.

The licensee's policy on replacing blown fuses will provide protection for fuse coordination. The effectiveness of this policy will be reviewed during subsequent inspections.

(b) High Impedance Fault Analysis

The high impedance fault concern is found in the case where multiple high impedance faults exist as loads on a safe shutdown power supply and cause the loss of the safe shutdown power supply prior to clearing the high impedance fault. Since the licensee's procedures to manually coordinate electrical circuits will provide protection for the high impedance fault concern, the licensee's protection for high impedance faults was found to be satisfactory.

(2) Spurious Signals

(a) High/Low Pressure Interfaces

High/low pressure interfaces are examined to determine if the licensee has provided measures to prevent fire induced spurious signals from producing a fire induced loss of coolant accident (LOCA). NRC guidance for protecting high/low pressure interfaces includes:

- Multiple (unlimited) hot short circuits, open circuits, and short circuits to ground are credible (the single spurious signal criteria does not apply).

- Three phase hot short circuits are credible.

- Hot short circuits in ungrounded DC circuits are credible.

The above guidance was employed in the review of the high/low pressure interface spurious signal concern at the Dresden Nuclear Station.

The licensee identified the shutdown cooling system on Units 2 and 3 as being high/low pressure interfaces. The licensee's analysis demonstrated protection for the Unit 2 and 3 shutdown cooling high/low pressure interfaces. Appendix C, shutdown cooling system high/low pressure interface protection, provides the technical details pertaining to the review of the Dresden Units 2 and 3 shutdown cooling high/low pressure interfaces.

(b) Isolation of Fire Instigated Spurious Signals

The licensee has provided isolation for fire instigated spurious signals by various methods, including:

- Administrative controls
- Isolation/transfer switches
- Fire wrap
- Cable relocation
- Manual component operation

The licensee has requested exemption for hot shutdown repairs to accomplish the following:

- To allow the pulling of fuses in order to place the condensate transfer pumps into local control.

- To allow the pulling of fuses to defeat high impedance faults.

- To allow the pulling and replacement of fuses on selected control circuits in lieu of redundant fusing.

The licensee's methodology of pulling fuses is considered a hot shutdown repair which is not permitted by Appendix R. The licensee had previously submitted an exemption request for fuse pulling. This is considered an Unresolved Item (237/88010-02(DRS); 249/88012-02(DRS)) pending disposition of the licensee's exemption request.

(3) Common Enclosure

The common enclosure associated circuit concern is found when redundant circuits are routed together in a raceway or enclosure and they are not electrically protected, or fire can destroy both circuits due to inadequate fire protection means.

During the inspection, licensee representatives stated:

- Redundant safe shutdown cables are never routed in common enclosures.

- Non safety-related cables may be routed in common enclosures with safety-related cables, but non safety-related cables are never routed between redundant safety-related divisions or trains.

- All cables are electrically protected.

During the inspection, randomly selected non safety-related cables routed in common enclosure with safety-related cables were verified to be electrically protected.

f. <u>Fire Instigated Spurious Operation of Unit 3 Target Rock Valve and Electromatic Relief Valve</u>

The NRC electrical inspector identified that, in the event of a disabling fire two hot shorts in a multiple conductor cable would cause spurious opening of the target rock valve and the electromatic relief valves. The licensee indicated that, based on Generic Letter 86-10 and discussion held with NRR, they had analyzed and provided protection for spurious operation of the target rock valve or one of the electromatic relief valves.

During the inspection the Appendix R inspection team consulted with NRR and were advised that the target rock valve and the electromatic relief valves were not considered to be high/low pressure interfaces. The NRR Technical Reviewer was informed of the potential simultaneous spurious operation of the target rock valve and all of the electromatic relief valves by failure of control cable 33934. The NRR Technical Reviewer stated that if failure of a single cable could cause the simultaneous spurious operation of more than one single electromatic relief valve, than a safety concern may exist. Further discussions between the NRR Technical Reviewer and the electrical inspector identified control cable 33934 as being a potential cable separation or common enclosure concern.

The inspector review of control circuits for Target Rock Valve 203-3A and Electromatic Relief Valves 203-3B, C, D, and E identified the following:

- The Target Rock Valve and Electromatic Relief Valves open when 125VDC power is supplied to the Target Rock solenoid or the respective Electromatic Relief Valve pickup coil.

- 125VDC power is supplied to the Target Rock Valve or Electromatic Relief Valves via the following relay contacts (Note: the listed relay contacts are installed in series with the respective solenoid or pickup coil):

VALVE	RELAY	CONTACTS	RELAY	CONTACTS
203-3A	2203-32/287-106B	7&8	2203-32/287-107B	8&7

203-3B	2203-32/287-106B or 2203-32/287-106A	9&10 5&6	2203-32/287-107B or 2203-32/287-107A	10&9 6&5
203-3C	2203-32/287-106B or 2203-32/287-106A	3&4 7&8	2203-32/287-107B or 2203-32/287-107A	4&3 8&7
203-3D	2203-32/287-106A or 2203-32/287-106B	9&10 5&6	2203-32/287-107A or 2203-32/287-107B	10&9 6&5
203-3E	2203-32/287-106A or 2203-32/287-106B	11&12 11&12	2203-32/287-107A or 2203-32/287-107B	12&11 12&11

- When positive 125VDC is applied to terminal 13 of relays 2203-32/287-106A, 2203-32/287-106B, 2203-32/287-107A, and 2203-32/287-107B the relays actuate to close the contacts listed above.

- Control Cable 33934, a 12-conductor 14 AWG cable, has conductors connected to terminal 13 of the above listed relays. One of the conductors in cable 33934 has positive 125 VDC applied from panel 903-32, terminal EE-21 (Drawing 12E-3462 SH 2 refers).

- Since Control Cable 33934 is installed downstream of the Auto Blowdown Inhibit Switch 903-3/287-304 contacts, auto blowdown inhibit may be bypassed by fire induced hot shorts in control cable 33934.

- Control Cable 33674, a multiconductor cable, has conductors connected to 2203-32/287-107B contact 11 via 2203-32 terminal BB-50 and 2203-32/287-107B contact 5 via 2203-32 terminal BB-39. One of the following spurious operations may occur from fire induced failure of Control Cables 33674 and 33934:

  - Hot shorting one conductor to the positive 125 VDC conductor in cable 33674 may cause either valve 203-3D or valve 203-3E to spuriously open.

  - Hot shorting two conductors to the positive 125 VDC conductor in cable 33674 may cause both valve 203-3D and valve 203-3E to spuriously open.

  - Hot shorting two conductors to the positive 125 VDC conductor in cable 33934 may cause valves 203-3A, B, C, D, and E to spuriously open.

The licensee's analysis indicated that the resolution for Control Cable 33674 discrepancies was: "Target Rock (manual function) and safety valves are available for RPV pressure control". The stated resolution does not demonstrate protection for simultaneous spurious opening of valve 203-3D and valve 203-3E.

The licensee's analysis also indicated that the resolution for Control Cable 33934 discrepancies was:

> "While in hot shutdown, it is necessary to prevent the electromatic relief valves from spuriously opening to preserve the reactor vessel coolant inventory. For fires external to the main control room, an AUTO BLOWDOWN INHIBIT switch at the MCB will prevent spurious blowdown. If the fire is in the MCB. it may be necessary to trip all of the power feeds to the blowdown logic. This is covered by procedures. Excessive reactor pressure will be controlled by the mechanically actuated target rock or safety valves."

This resolution does not appear correct since fire induced failures of cables 33674 and 33934 may bypass and defeat the function of the Auto Blowdown Inhibit Switch at the Main Control Board. The licensee's resolution to trip all of the power feeds to the blowdown logic being covered by procedures is correct; however, the implemented procedures were developed to provide protection for the spurious opening of either the Target Rock Valve or one of the Electromatic Relief Valves. Since simultaneous spurious opening of the Target Rock Valve and all of the Electromatic Relief Valves has not been analyzed, procedural protection has not been demonstrated.

The simultaneous spurious opening of the Target Rock Valve and Electromatic Relief Valves has a tremendous impact on reactor coolant inventory based on the limited capacity of the CRD Hydraulic System to restore or maintain reactor coolant inventory. Due to the significance of this issue and its generic implications, the spurious operation of the Target Rock Valve and Electromatic Relief Valves has been referred to NRR. This is considered an Unresolved Item (237/88010-03(DRS); 249/88012-03(DRS)) pending resolution from NRR.

On August 15, 1988, the inspector met with the licensee to discuss appropriate fire protection features and measures to prevent or mitigate consequences or spurious operation of the Target Rock Valve and Electromatic Relief Valves. In addition, the inspector walked down the areas of concern. As a result of the discussions and walk down of the areas on August 15, 1988, a conference call was conducted on August 17, 1988, between Dresden, Quad Cities, CECo Licensing and Region III to discuss the fire protection features and compensatory measures that would be taken to prevent or mitigate the consequence of a disabling fire from causing a spurious operation of the Target Rock Valve and Electromatic Relief valves.

Attachment D of a September 16, 1988 letter from J. Silady, CECo, to T. Murley, NRC, summarized the conference call of August 17, 1988. In this letter, the licensee indicated that in all areas through which the subject cables are routed, there are automatic suppression and detection systems, except for the mezzanine floor of the Dresden Unit 3 Reactor Building which only has a detection system. The following Interim Compensatory Measures were implemented for the affected areas of the reactor building mezzanine floor:

- Declared area combustible free fire zone
- Additional portable fire fighting equipment was brought into the area
- A combustible loading inspection was conducted per shift basis by station operators

The licensee indicated that the above actions would commence immediately and be in effect until this issue is resolved.

g. Availability of Opposite Unit Safe Shutdown Equipment During Refueling Outages

The licensee has selected two primary systems for achieving hot shutdown in the event of a disabling fire concurrent with a loss of offsite power. The systems are the isolation condenser system and the HPCI system. As previously mentioned, five different Appendix R hot shutdown paths per unit are identified in the licensee's safe shutdown methodology. Four of the paths per unit utilize the respective unit's isolation condenser, and differ only in that they employ different power trains, diesel generators, CRD pumps, and/or operating methods. The fifth path per unit is the HPCI/LPCI method of shutdown.

The Dresden safe shutdown procedures utilize safe shutdown equipment from the unaffected unit during certain fire scenarios. Included in this equipment are condensate transfer pumps, CRD pumps, service water pumps, 4KV and 480V busses, and 480V MCCs. It was identified by the inspectors that the licensee had no administrative controls to insure that the required opposite unit equipment was available or that compensatory measures would be in place during a refueling outage to insure that at least one train of safe shutdown equipment was available in the event of a disabling fire in the operating unit.

The failure of the licensee to establish procedures or controls to ensure that required alternative shutdown equipment was available for safe shutdown of the operating unit when the opposite unit (which houses the alternative shutdown equipment) was in an outage or shutdown and the required alternate shutdown equipment was removed from service for scheduled maintenance or repair is considered a violation of Appendix R to 10 CFR 50 (237/88010-05(DRS); 249/88012-05(DRS)) as described in the Notice of Violation.

At the request of the inspectors, the licensee developed draft administrative controls for safe shutdown equipment during refueling outages (letter dated April 21, 1988, from E. D. Eenigenburg, CECo, to J. Holmes, NRC).

The draft administrative procedure has been forwarded to NRR for review.

h. Fire Protection of Safe Shutdown Capability

In the licensee's safe shutdown report, the licensee has identified several safe shutdown pathways in which at least one pathway per unit will be available in the event of a disabling fire in either Unit 2 or Unit 3. The inspectors toured both units and observed fire walls and suppression and detection systems which appeared to be well designed and installed as described in the Safe Shutdown Report. There were no identified discrepancies, however, a concern has been identified regarding the adequacy of separation of Unit 1 from Unit 2/3 (See Unresolved Item (237/88010-01(DRS); 249/88012-01(DRS)) of this report).

4. Fire Protection Features

As part of the Appendix R compliance assessment, several fire protection features were also reviewed as listed below:

- Carbon Dioxide Systems
- Control of Combustibles

a. Carbon Dioxide System

The licensee has provided total flooding carbon dioxide (CO_2) suppression systems for the AEER, three diesel generators and the Diesel Tank Rooms.

The inspector requested the original CO_2 concentration test results for the diesel generator rooms. The licensee indicated that the original tests were not available however CO_2 concentration tests were planned to be conducted by the end of the Unit 2 refueling outage. At the request of the inspector, the licensee performed puff tests on Diesel Generator No. 3.

As a result of the test, the licensee was informed of the following inspector observations:

(1) Two employees entered the testing area immediately after the CO_2 discharge test. Measures should be provided to ensure that personnel will not enter the test area until the appropriate personnel have tested the area to ensure it is safe to enter.

(2) Procedures should inform test personnel of specific fire dampers and other equipment that are expected to function during the performance of the test.

(3) During the CO_2 puff test, Damper 3-5772-102 failed to close.

(4) The predischarge alarm for the diesel room CO_2 system is an audible alarm. There is no visual alarm. The licensee was requested to verify that the audible alarm is sufficient to warn personnel that may be in the area with the diesel operating. This is considered an Open Item (237/88010-05(DRS); 249/88012-05(DRS)) pending review of the licensee's actions.

As a result of the CO_2 auxiliary equipment failure the licensee initiated a work request for HVAC Damper 3-5772-102. The licensee acknowledged the inspector's concerns previously identified and indicated that actions regarding these concerns would be tentatively completed by November 30, 1988.

b. Control of Combustibles

10 CFR 50.48(a) requires that each operating nuclear power plant have a fire protection plan that satisfies Criterion 3 of Appendix A to 10 CFR Part 50. It further requires that the plan describe specific features necessary to implement the program such as administrative controls to limit fire damage to structures, systems, or components important to safety so that the capability to safely shutdown the plant is ensured.

The licensee satisfied Criterion 3 by meeting the applicable requirements of Sections III.G, III.J, and III.O of Appendix R and by meeting the fire protection requirements identified in the guidelines of Appendix A to Branch Technical Position B.T.P APCSB 9.5-1 as reflected in the staff fire protection safety evaluation issued prior to the effective date of the Appendix R rule.

In Section B.2 of the licensee's response to the guidelines of Appendix A to APCSB 9.5-1 (Amendment 2-2/86) the licensee indicated that effective administrative measures have been implemented to prohibit bulk storage of combustible materials inside or adjacent to safety-related buildings or systems during operation or maintenance periods.

The licensee has implemented Administrative Procedure "Control of Transient Combustibles, Storage Areas and No Smoking Areas." The procedure establishes guidelines for storage and handling of transient combustible materials in certain areas of the plant which contain safety-related components and/or equipment important to safe shutdown. These plant areas are identified below:

- Unit 2/3 Reactor Building
- Unit 2/3 Turbine Building
- Unit 2/3 Cribhouse
- Unit 1 Cribhouse

The procedure is provided with a fire loading of common material chart that establishes a low, medium or high fire load based on the amount and type of material (such as wooden scaffolding, Class 1 or 2 combustible liquids, etc).

The procedure indicates that for low transient fire loads, additional fire protection equipment or a work permit is not required. However, the work area should be kept clean and all materials removed as soon as practicable.

For medium transient fire loads, the procedure requires the completion of a transient combustible permit (DAP Form 3-3A) and the placement of a supplemental fire extinguisher at the jobsite. The permit is to be completed by the responsible work group supervisor or foreman and posted in the work area. One copy should be sent to the Fire Marshal. The procedure further indicates that the Fire Marshall may specify additional requirements upon receipt of the permit.

For high transient fire loads the responsible work group supervisor must obtain the approval of the Fire Marshal on the permit before posting it in the work area. At this time, the Fire Marshal will specify appropriate requirements on the permit such as supplemental fire extinguishers, hoses or fire watches.

The procedure further specifies that during major outages, transient combustibles shall be controlled in accordance with the combustible control procedure except that the accumulation of transient combustibles is permitted provided the accumulation of transient combustibles does not exceed that which could be removed by the end of the next normal shift.

On April 12, 1988, when Unit 2 was operating and Unit 3 was in an outage, an NRC inspector identified that approximately twenty 55-gallon drums of lubricating (lube) oil was stored in a safety-related area on Elevation 517' - 6" (southwest corner - Fire Area 1.1.2.2) of the Unit 2 Reactor Building. The licensee had previously requested an exemption from Section III.G.3 of Appendix R that required fixed fire suppression system in this area.

The licensee indicated to the inspector that on March 31, 1988, the lube oil was transferred to the Unit 2 side for approximately 13 days before it was discovered as a problem and transferred to the Unit 3 trackway. The inspector requested the transient combustible permit for the lube oil observed at the Unit 2 Reactor Building, Elevation 517'-6", Southwest corner. The licensee was unable to provide the inspector with the transient combustible permit (short term document) for the specified storage of the lube oil in the Unit 2 Reactor Building. However, according to the Fire Marshal, he was informed of the transfer of lube oil to the Unit 2 side and concluded that the temporary storage of lube oil was acceptable and no additional fire protection features were required based on the following:

(1) Low traffic area
(2) Fire detection was available
(3) Lube oil flash point characteristics were such that it was difficult to ignite
(4) Safe shutdown could have been achieved in the event of a disabling fire utilizing the equipment for safe shutdown Path B-1

The licensee had established a formal transient combustible procedure however it did not appear that the transient combustible permit was effective for preventing the storage of lubricating oil in the Unit 2 Reactor Building. The failure of the licensee to meet the requirements of their approved fire protection program by permitting the storage of twenty 55-gallon drums of lube oil in the safety-related area is considered a violation (237/88010-06(DRS); 249/88012-06(DRS)) as described in the Notice of Violation.

5. Open Items

Open items are matters that have been discussed with the licensee, that will be reviewed further by the inspector, and that involve some action on the part of the NRC, the licensee, or both. Open items disclosed during the inspection are discussed in Paragraphs 3.e and 4.a.

6. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, or items of noncompliance, or deviations. Unresolved items disclosed during the inspection are discussed in Paragraphs 2.e, 3.e, 3.f, 3.g, 4.a and 4.b.

7. Exit Meeting

The inspector met with licensee representatives on October 18 and December 13, 1988. The licensee indicated the likely content of this report and the information discussed during the inspection was not considered proprietary in nature.

NRC Appendix R Audit Questions April 21, 1988 at 15:38:29

No.	Subject	Asked By	Responsible Engrs	Response ~~Received~~	Resolution Comt Date
1	Operation of iso switches tour/demo of action and fuse pulling	KP	RW, BMB	X	NA
2	Provide of fire area/ path drawings	JH	JK	X	NA
3	Why oil barrel without permit	JH	MD	X	NA
4	Inventory for DG 2/3 SSD Box	KP/RH	RJ	X	NA
5	Sprinkler orientation	JH	CK, MD	X	NA
6	Inadvertent CO2 discharge	KP	BMB	X	NA
7	Maintence procedures and schedules for bkrs and relays	KP	RW	X	NA
8	Calc for reactor water makeup	RH	BMB	X	NA
9	Sign-off for EPIP 200-20	RH	RJ	X	NA
10	SSD equip OOS controls on unaffected unit	RH	BMB		
11	Common bus, CT circuits, high impedence faults	KP	FWF, CER	X	NA
12	Open items from 1985	JH	WP		
13	Control of fuse replacement	KP	RS	X	NA
14	Breaker coordination curves	KP	FWF	X	NA
15	Provide fire stop detail	KP	RW	X	NA
16	High/low interface cable routing and control proc	KP	REB	X	NA
17	Provide cable routings	KP	REB, FWF, CER	X	NA
18	See detector dsgn abv 24-1	JH	MD		

NRC Appendix R Audit Questions April 21, 1988 at 15:38:49

No.	Subject	Asked By	Responsible Engrs	Resolved	Comt Date
19	Hatchway draft curtain	JH	CK, MM	X	NA
20	Penetration seal labels	JH	BMB	X	NA
21	Fire Pump Tests	JH	MD		
22	Light blocks detectors	JH	RB, SW, RL	X	NA
23	Emergency light tests	JH	MD		
24	CO2 puff test	JH	MD		
25	Cal histories on breakers and relays	KP	JJ, RS	X	NA
26	Maint Surveillance Freq breakers	KP	JJ	X	NA
27	87-50 concerns on test-able check valves	KP	JFK, CML	X	NA
28	Fire barrier surviel-lance	JH	MD, CK		
29	Penetration seal	JH	MD, CK		
30	Availability of 125-Vdc	KP	CER, FWF, RW, BMB	X	5/3/88
31	250-Vdc battery chargers	KP	CER	X	NA
32	Function of device on side of SW pumps-oil water measurement	JH	BMB	X	NA
33	Exemption request for cribhouse fire detection	JH	CML	X	NA
34	Effect of water spray on electrical cables	JH	CER	X	NA
35	Spurious operation of valve 1001-20: high/low pressure interface	KP	BMB, REB, CER	X	NA
36	Review of mod review process	KP	BMB	X	NA

NRC Appendix R Audit Questions April 21, 1988 at 15:39:11

No.	Subject	Asked By	Responsible Engrs	Resolved	Comt Date
37	Exemption request for DG 2/3 fuse replacement	KP	CML	X	NA
38	Common enclosure	KP	CER, FWF	X	NA
39	Teledyne info on yearly discharge tests	JH	BMB		
40	Upgrade DFPP 4153-2 to include mfgr instructions on disc level	JH	BMB		
41	Provide last monthly inspec-	JH	BMB	X	NA
42	ERV-high/low interface	KP	JFK, CML	X	NA
43	Pump curve for Unit 1 fire	JH	MM, MD, CK		
44	Provide copy of transient combustible permit for Unit 3 HPCI oil drums	JH	MD		
45	Unplug Emergency Lights	JH	MD		
46	Tech spec manning require-	RH	BMB	X	NA
47	Current transformers	KP	CER	X	NA
48	Wrapping trays	KP	CER	X	NA
49	Division II tray junction with Division I	KP	CER	X	NA

DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4-19-88

CE CO./AE PERSON QUESTIONED: C. Ruth / F. Fischer

COGNIZANT NRC PERSON K. Parkinson

LOCATION/AREA WHERE QUESTION WAS ASKED: Auditorium East (NRC area)

QUESTION/ITEM DISCUSSED: Operation of isolation switches at breakers: Requested tour to observe operator action required. Also removal of control power fuses.

FOLLOW-UP ACTIONS: PROVIDE TOUR Scheduled for
TOUR: BRIAN BARTH / OPERATOR

RESOLUTION: Keith Parkinson asked operator to open IS compartment during walkdown - acceptable

PERSONNEL INVOLVED IN RESOLUTION: R. Whalen
B. Barth

FOLLOW-UP ACTION IF ANY:

(2)

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DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4/18/88

CE CO./AE PERSON QUESTIONED: R. Wikalow

COGNIZANT NRC PERSON J. Holmes

LOCATION/AREA WHERE QUESTION WAS ASKED: U2 RB 517

QUESTION/ITEM DISCUSSED: NEEDS SET OF DRAWINGS SHOWING FIRE AREAS / SSD PATHS

FOLLOW-UP ACTIONS: PROVIDE TO HOLMES

RESOLUTION: Drawing were provided 4-19-88 (Am)
SSD REPORT VOL. 3 BOOK 1 Figure 2.2-1 2.23
AND TABLE 2.1-1.

PERSONNEL INVOLVED IN RESOLUTION: J. Kelly

FOLLOW-UP ACTION IF ANY:

DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4/18/88

CE CO./AE PERSON QUESTIONED: M. DILLON

COGNIZANT NRC PERSON J. HOLMES

LOCATION/AREA WHERE QUESTION WAS ASKED: 2/3 DG ROOM

QUESTION/ITEM DISCUSSED: WHY OIL WITH NO PERMIT (BARREL)
NO COMMITMENT? LOOK IT UP.

FOLLOW-UP ACTIONS: RESPOND 55 gallon Drum will be removed by afternoon of 4-19-88

RESOLUTION: 55 GALLON DRUM REMOVED 4/19/88 WA

PERSONNEL INVOLVED IN RESOLUTION: M. DILLION

FOLLOW-UP ACTION IF ANY:

(4)

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DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4/18/88

CE CO./AE PERSON QUESTIONED: B. BATH

COGNIZANT NRC PERSON K. PARKINSON / RUDY

LOCATION/AREA WHERE QUESTION WAS ASKED: 2/3 DG -

QUESTION/ITEM DISCUSSED: INVENTORY FOR 2/3 DG SSD BOX - WHY NO FUSE PULLERS? DO WE NEED THEM? COPY OF SURVEILLANCE -

FOLLOW-UP ACTIONS: M. DILLON CALLED FOR REPL FUSE PULLERS IMMEDIATELY - PROVIDE COPY OF SURVEILLENCE

RESOLUTION: DOS 010-14 GIVEN TO RUDY/ KEITH.

PERSONNEL INVOLVED IN RESOLUTION: R. JOHNSON

FOLLOW-UP ACTION IF ANY: D Calhoun and E Skowron walked down safe shutdown boxes.

(5)

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DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22,1988

DATE: 4/18/88

CE CO./AE PERSON QUESTIONED: P. WHALEN

COGNIZANT NRC PERSON J. HOLMES

LOCATION/AREA WHERE QUESTION WAS ASKED: U2 RB 2ND Floor

QUESTION/ITEM DISCUSSED: WHY IS SPRINKLER ABOVE SDC PP ROOM - CHASE INCLINED. - KSOBIECH

FOLLOW-UP ACTIONS: FIRE PROTECTION ENGINEER REVIEWED AREA. PROTECTION FOR PIPE CHASE IS O.K. EVEN WITH PIPE INCLINED. FOR SYSTEM ENHANCEMENT WORK REQUEST TO BE WRITTEN TO ADJUST HANGER & STRAIGHTEN PIPE. 74504 WRITTEN 4-19-88.

RESOLUTION:

PERSONNEL INVOLVED IN RESOLUTION: KSOBICH

M. DILLON

FOLLOW-UP ACTION IF ANY:

RR30

DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4/18/88

CE CO./AE PERSON QUESTIONED: M. DILLON

COGNIZANT NRC PERSON KEITH PARKINSON

LOCATION/AREA WHERE QUESTION WAS ASKED: CARDOX TANK

QUESTION/ITEM DISCUSSED: CAN CENTRAL AREA FIRE CAUSE discharge of CO_2 into DG rooms

FOLLOW-UP ACTIONS: Presented S&L analysis for CO_2 panel letter dated November 17, 1987

RESOLUTION: OK - 4-19-88 (BGB)

PERSONNEL INVOLVED IN RESOLUTION: BB Booth

FOLLOW-UP ACTION IF ANY:

7

DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4/18/88

CE CO./AE PERSON QUESTIONED: C. Ruth

COGNIZANT NRC PERSON K. Parkinson

LOCATION/AREA WHERE QUESTION WAS ASKED: Auditorium

QUESTION/ITEM DISCUSSED: Maintenance procedures and schedule(s) for breakers and relays are requested.

FOLLOW-UP ACTIONS: JURECKI TO DISCUSS TUES. ~~AFTERNOON~~.

RESOLUTION: Discussed 4-19-88

PERSONNEL INVOLVED IN RESOLUTION: R. Whalen

FOLLOW-UP ACTION IF ANY:

DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22,1988

DATE: 04-19-88

CE CO./AE PERSON QUESTIONED: B Barth

COGNIZANT NRC PERSON R Hada

LOCATION/AREA WHERE QUESTION WAS ASKED: Conference Room

QUESTION/ITEM DISCUSSED: calculation for ~~opening opening of EMRV~~ TIMELINE TO TOP OF CORE

FOLLOW-UP ACTIONS: PROVIDE GE calculation

RESOLUTION: PROVIDED ON 04-19-88
VOLUME 6, BOOK 2,

PERSONNEL INVOLVED IN RESOLUTION: B.M.B

FOLLOW-UP ACTION IF ANY:

DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22,1988

DATE: 4-19-88

CE CO./AE PERSON QUESTIONED: B Bach

COGNIZANT NRC PERSON R Hodor

LOCATION/AREA WHERE QUESTION WAS ASKED: Auditorium

QUESTION/ITEM DISCUSSED: sign off for EPIP 200-20

FOLLOW-UP ACTIONS: presented to R Hodor on 4-19-88

RESOLUTION: acceptable BB 4/19/88

PERSONNEL INVOLVED IN RESOLUTION: R. Johnson

FOLLOW-UP ACTION IF ANY:

DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4/18/88

CE CO./AE PERSON QUESTIONED: C. Ruth

COGNIZANT NRC PERSON K. Parkinson

LOCATION/AREA WHERE QUESTION WAS ASKED: Auditorium

QUESTION/ITEM DISCUSSED: Be ready to discuss the following tomorrow:
1. Common Bus Concern for selected buses
2. CT circuits
3. High Impedance Faults

FOLLOW-UP ACTIONS: Referred to the following sections of Vol. 3 Bk. 1:
1. Pages 3.3-4 and 5.5-1
2. Page 5.3-1
3. Page 5.5-1

RESOLUTION: See subsequent questions on these topics.

PERSONNEL INVOLVED IN RESOLUTION: FISCHER
RUTH

FOLLOW-UP ACTION IF ANY: Documented separately

DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4-19-88

CE CO./AE PERSON QUESTIONED: R. Stachniak

COGNIZANT NRC PERSON Keith Parkinson

LOCATION/AREA WHERE QUESTION WAS ASKED: Auditorium Meeting Room

QUESTION/ITEM DISCUSSED: Fuse replacement: Provide a station procedure - for controling the replacement of fuses - How do we assure that proper fuses are replaced?

FOLLOW-UP ACTIONS: Keith was shown monthly surveillance of Appendix R fuse checks - If ~~a~~ Appendix R cables are in a common enclosure with non- Appendix R ~~fuse~~ cables, the surveillance is not sufficient.

RESOLUTION: Keith was given copy of attached letter as resolution

PERSONNEL INVOLVED IN RESOLUTION:

R Johnson
D. Strobel
J. Kotowski

FOLLOW-UP ACTION IF ANY:

April 20, 1988

JK LTR: #88-019

To: Operations Department Personnel

Subject: Policy on Replacing Blown Fuses

A procedure will be prepared to formalize the Operations Department policy on replacing blown fuses. That policy is as follows:

1. Compare the new fuse to the old fuse to verify that they are "like for like." This comparison should include manufacturer, physical size, shape, voltage and current ratings, and fuse type (quick-acting, slow-blow, etc.).

2. If illegible or missing markings for the old fuse do not permit a complete verification of voltage and current ratings or fuse type, the Shift Supervisor will obtain verification of such data from wiring diagrams and/or vendor manuals or by consultation with the Technical Staff.

Joe Kotowski
Asst. Supt. - Operations

JK:RJ:rg

cc: J. Wujciga
Operating Engineers

DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22,1988

DATE: 4-19-88

CE CO./AE PERSON QUESTIONED: FRED FISCHER

COGNIZANT NRC PERSON KEITH PARKINSON

LOCATION/AREA WHERE
QUESTION WAS ASKED: ADMIN AUD.

QUESTION/ITEM
DISCUSSED: ARE AC POWER SYSTEM COORDINATION CURVES AVAILABLE? AC COORDINATION WAS NOT USED TO ADDRESS COMMON POWER SUPPLY, MANUAL ACTIONS TO OPEN BREAKERS WAS USED. MR PARKINSON SAID THAT IS ACCEPTABLE FOR APPENDIX R CONCERNS, BUT BREAKER COORDINATION IS A SEPARATE TOPIC. FWF SAID THIS WAS UNDER DISCUSSION WITH THE REGION AS AN OUTCOME OF THE QUAD AUDIT.

FOLLOW-UP
ACTIONS: NONE

RESOLUTION:

PERSONNEL INVOLVED IN
RESOLUTION: FRED FISCHER

FOLLOW-UP ACTION IF ANY:

DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4/18/88

CE CO./AE PERSON QUESTIONED: B. Wilson

COGNIZANT NRC PERSON Keith Parkinson

LOCATION/AREA WHERE QUESTION WAS ASKED: U2 RB 517

QUESTION/ITEM DISCUSSED: WOULD LIKE DRAWINGS FOR FIRE STOPS (PANS IN OVERHEAD) —
PERHAPS DFPP SURVEILLANCE OR F-DRAWINGS CONCEPT REVIEW WOULD ANSWER HIS CONCERN —

FOLLOW-UP ACTIONS: BMB provided 12E6508A drawing showing gypsum fire break

RESOLUTION: OK BMB 4/19/88

PERSONNEL INVOLVED IN RESOLUTION: B. BARTH

FOLLOW-UP ACTION IF ANY:

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DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4-19-88

CE CO./AE PERSON QUESTIONED: C.E. RUTH/R.E. BROWN

COGNIZANT NRC PERSON K. PARKINSON

LOCATION/AREA WHERE QUESTION WAS ASKED: NRC AREA - EAST AUDITORIUM

QUESTION/ITEM DISCUSSED: HIGH-TO-LOW PRESSURE INTERFACE:
REQUESTED CABLE ROUTING FOR VALVE 3-1001-2C
PROCEDURE TO CONTROL HIGH/LOW INTERFACE

FOLLOW-UP ACTIONS: PROVIDED HIGHLIGHTED PRINTS OF DRAWINGS
12E-20A8 thru 2053 (ROUTING), 3904D, E, 3905M (CABLE TABS)

RESOLUTION: Provided 4-~~20~~ 19-88

PERSONNEL INVOLVED IN RESOLUTION: R.E. BROWN

FOLLOW-UP ACTION IF ANY: NONE

RR30

DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4/19/88

CE CO./AE PERSON QUESTIONED: R. Stachniak

COGNIZANT NRC PERSON Keith Parkinson

LOCATION/AREA WHERE QUESTION WAS ASKED: Auditorium Meeting Room

QUESTION/ITEM DISCUSSED: Keith would like to walk down cables to spot check routing

20682 } 23936 30659 } service water
22869 } ~~22934~~ 23935 30766 } pump
23904
20794 } 23927 ~~[illegible]~~ D/G excitation
22780 } 25065 schematic dwgs
Need cable routing for each

20271 Target rock & electromatic cables

FOLLOW-UP ACTIONS: DRAWING MARKED UP + GIVEN TO MR PARKINSON ON 4/19 + 4/20

RESOLUTION: SEE ABOVE

PERSONNEL INVOLVED IN RESOLUTION:
R. E. Brown
F. W. Fischer
C. E. Ruth

FOLLOW-UP ACTION IF ANY:

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DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4/19/88

CE CO./AE PERSON QUESTIONED: M. DILLON

COGNIZANT NRC PERSON J. HOLMES

LOCATION/AREA WHERE QUESTION WAS ASKED: 3 REACTOR BUILDING.

QUESTION/ITEM DISCUSSED: NEED TO DISCUSS DRAFT CURTAIN FOR HATCHWAY SPRINKLERS AND STAIRS. WANTS TO TALK TO PLC.

FOLLOW-UP ACTIONS:

RESOLUTION: provided National Bureau of Standards reference which found that draft curtains did not materially affect sprinkler performance when protecting stair openings (4/20 AM)

PERSONNEL INVOLVED IN RESOLUTION: CHRIS KSOBIECH
MIKE MOWRER

FOLLOW-UP ACTION IF ANY: NONE

DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4/19/88

CE CO./AE PERSON QUESTIONED: M. DILLON

COGNIZANT NRC PERSON J. HOLMES

LOCATION/AREA WHERE QUESTION WAS ASKED: U2 REACTOR BLDNG

QUESTION/ITEM DISCUSSED: ARE ALL PENETRATIONS GOING TO BE LABELLED SIMILAR TO PENETRATIONS THAT ARE LABELLED.
SCRD ROOM PENETRATIONS LABELLED
EL 545 WEST WALL TO U3 3 LARGE PIPES NOT LABELLED.
EL 545 NORTH WALL BY ELEVATOR NOT LABELLED.

FOLLOW-UP ACTIONS: all accessable penetrations will be labeled. Penetrations in floor may be labeled in a different manner than the walls

RESOLUTION:

PERSONNEL INVOLVED IN RESOLUTION: B Barth

FOLLOW-UP ACTION IF ANY:

DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4/19/88

CE CO./AE PERSON QUESTIONED: M. DILLON

COGNIZANT NRC PERSON J. HOLMES

LOCATION/AREA WHERE QUESTION WAS ASKED: 2 REACTOR BLDNG

QUESTION/ITEM DISCUSSED: LIGHT FIXTURES BLOCKING SMOKE DETECTOR (2-4131-225)

CABLE BY ROLL-UP FIRE DOOR (U2 TO U3) 517' MAY BLOCK DOOR FROM CLOSING.

FOLLOW-UP ACTIONS: CABLE WAS MOVED AWAY FROM DOOR (FIRE DOOR 57 WR#74502.)

RCT TO SURVEY LIGHT FIXTURE AREA EM'S WILL THEN REHANG ONE LIGHT IN PROPER LOCATION AND REMOVE THE OTHER LIGHT.

RESOLUTION:

PERSONNEL INVOLVED IN RESOLUTION: R. BLACK - ASST. FIRE MARSHAL
S. WEBER - E.M. FOREMAN
R. LEE - RCT

FOLLOW-UP ACTION IF ANY:

DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4/19/88

CE CO./AE PERSON QUESTIONED: M. DILLON

COGNIZANT NRC PERSON J. HOLMES

LOCATION/AREA WHERE
QUESTION WAS ASKED: AUDITORIUM

QUESTION/ITEM
DISCUSSED: EMERGENCY LIGHTING - CONDUCT AN EMERGENCY LIGHTING TEST (8 HR TEST) ≈ 6 LIGHTS. (See sheet 43) THURSDAY (AM).

1. PROVIDE SURVEILLANCE PROCEDURES.
2. MANUFACTURERS INSTRUCTIONS
3. BLACK-OUT TEST.

FOLLOW-UP
ACTIONS: M. DILLON / R. BLACK TO SCHEDULE TEST
C. YSOBIECH TO LOOK OF PROCEDURES / MFR'S INFO / ADEQUACY WALKDOWN

RESOLUTION: ① Showed DFPP 4153-2 rev 5 dated February, 1988 ② Showed Manufacturers Instruction Manuals of Beam-O-Matic and Teledyne Big-Beam, and ③ Black-out Test from Vol 4, book 2 Section V to Mr. Holmes
Gave J.H. Teledyne Monthly checks list from Mfg literature

PERSONNEL INVOLVED IN
RESOLUTION: B Barth
E. Hunniut
J Kosiecek

FOLLOW-UP ACTION IF ANY: Open Item to be issued for:
1. Obtain Teledyne Mfg Instruction Manual
2. Establish a yearly inspection procedure for discharge, test what others did
3. Include check for hydrometer' disc per Mfg. data sheet
4. Revise procedure to record voltage listed during monthly inspections to allow for trend analysis on units.

RR30 ⑤ Revise procedure to ID the few units that have 12 V. with 1 panel it not specifically.

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DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4-19-88

CE CO./AE PERSON QUESTIONED: Fidel Marquez (OAD)

COGNIZANT NRC PERSON Keith Parkinson

LOCATION/AREA WHERE QUESTION WAS ASKED: Auditorium/Meeting Room

QUESTION/ITEM DISCUSSED: Wants calibration histories for relays on breakers feeding 2A CRD 2/3 DG diff. & 3 B service water pump.

FOLLOW-UP ACTIONS: Provided attached OAD calculation

RESOLUTION: acceptable BG 4/19/88

PERSONNEL INVOLVED IN RESOLUTION: Jerry Jurecki
Rob Stachniak

FOLLOW-UP ACTION IF ANY:

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DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4-19-88

CE CO./AE PERSON QUESTIONED: Jerry Jurecki (EMD)

COGNIZANT NRC PERSON Keith Parkinson

LOCATION/AREA WHERE QUESTION WAS ASKED: Auditorium/Meeting Room

QUESTION/ITEM DISCUSSED: Wanted maintenance surveillance frequency for breaker maintenance -

FOLLOW-UP ACTIONS: Jerry provided this to Keith

RESOLUTION:

PERSONNEL INVOLVED IN RESOLUTION: J Jurecki

FOLLOW-UP ACTION IF ANY:

RR30

REMARKS:

* RELAY REPLACED FOR WYLE LAB. MOD.

Rob

RELAY	MODEL OR STYLE	OTHER
CFD	12 CFD B1A	MIN. P.U. 0.2A

AUX. RELAY [redacted] TRIPPING 152-2333 (23-1 FD. BKR); 152-3333 (33-1 FD. BKR);

LOAD RES. DIESEL 2/3 FUEL

ALARM # 2212 INTERLOCK PRINT # 12E 2345

SETTINGS

MIN P.U.	0.2A						
% SLOPE							
TAP							
REL. TAPS							
TR. KVA							
TR. TAPS KV							
KV							
KV							
DATE:	[redacted]						

COPY

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EQUIP.	C.T. LOC.	C.T. TYPE & RATIO	SET	AUX.C.T.TYPE,RATIO	CORR. RATIO
BUS 23-1 FD BKR FROM DG 2/3	BUS SIDE BKR	600/5	120/1		
BUS 23-1 FD BKR FROM DG 2/3	BUS SIDE BKR	600/5	120/1		
D.G 2/3	NEUT. SIDE GEN	600/5	120/1		

TEST DATA		AS FOUND	AS LEFT						
MIN. OPER. WINDING AO	1 5-6	0.2	0.2						
	1 6-7	0.2	0.2						
	2 5-6	0.2	0.2						
	2 6-7	0.2	0.2						
	3 5-6	0.2	0.2						
	3 6-7	0.2	0.2						
% SLOPE									
AMMETER		0514055D							
W.R. #		D605-5							

AUX. RELAY	186 DG 2/3							
TRIP	✓							
DATE:	7-9-87							
BY:	[illegible]							

LOC. STA. 12 BUS 40 ~~[illegible]~~ EQUIP. ~~[illegible]~~ TO D.G. 2/3 DIFF CFO DIFF. RELAYS

COPY

QUIP.	C.T. LOC.	C.T. TYPE & RATIO	SET	AUX.C.T.TYPE,RATIO	CORR. RATIO
BUS 23-1 FD. FROM DC 4/5		600/5	120/1		

TEST DATA

OPER. WINDING	5-6 1 6-7	.204/.204	.205/.206	.200/.200	.195/.196	.200/.201	.201/.200		0.20/0.20
	5-6 2 6-7	.200/.200	.204/.203	.201/.200	.199/.195	.200/.201	.205/.205	0.20/0.20 *	0.20/0.20
	5-6 3 6-7	.200/.200	.204/.203	.203/.205	.198/.197	.200/.201	.201/.201		0.20/0.20
% SLOPE									
				AMMETER	258836			051455

AUX. RELAY	HEA	HEA		HEA		HEA		
TRIP	OK	OK	✓	✓		✓		OK
DATE:	10-20-74	2-15-77	4-13-78	4-9-79	3-11-81	2/9/85		[illegible]
	GKS-LRT	JWS LRT	JWS	MJJ	F CLS	E TK		[illegible]

DC. STA-12 BUS 23-1 EQUIP. BUS 23-1 FD. FROM D.C. 4/5 CFD DIFF. RELAYS

REMARKS:

RELAY	MODEL OR STYLE	RANGE
IAC 66	12IAC66B2A	20-80A (IUST) 2.5-5.0 (TOC)
PJC 11.	12PJC11AV1A	.5 - 2.0

AUX. RELAY TRIPPING CRD FEED PUMP 2A BKR (ACB152-2301

LOAD RES. ALARM MCR 12E-2416

SETTINGS	AØ CØ	I				
INST. RELAY A	50	0.5				
INST. LINE A	500	5.0				
TAP A.	5.0					
PRI.A.	50					
T.L.	3.0					
CURVE TEST	12.5A→20"					
DATE	12-21-86	12-21-86				
BY—	[illegible]	[illegible]				

COPY CURRENT & VOLTAGE RELAYS - NUCLEAR STA.

OPER. TYPE LOCATION LOAD SIDE OF BKR
RATIO 50/5 50/5 SET
AUX.C.T. LOC. RATIO SET
POL. TYPE LOC. RATIO

	LIST ALL CERTIFIED TEST INSTRUMENTS:		LIST ALL CERTIFIED TEST INSTRUMENTS:		LIST ALL CERTIFIED TEST INSTRUMENTS:		LIST ALL CERTIFI TEST INSTRUMENTS	
AMMETER	05140450							
TIMER	17540250							
INST AMP.	[illegible]							
	WORK REQUEST NUMBER: D[illegible]		WORK REQUEST NUMBER:		WORK REQUEST NUMBER:		WORK REQUEST NU	
	AS FOUND	AS LEFT	AS FOUND	AS LEFT	AS FOUND	AS LEFT	AS FOUND	AS
TEST	CO CO ±	AZ CO ±						
INST.A.	50 50 .5	50 50 .5						
TAP	5.0 5.0	5.0 5.0						
T.L.	3.8 3.4	3.8 3.4						
MIN.A.	5.0 5.0	5.0 5.0						
% TAP	3X	3X						
SEC.	16.9 [illegible]	[illegible]						
% TAP	5X	5X						
SEC.	11.1 11.3	[illegible] 11.3						
DIR. SEN.								
C.T.MAG.	w [illegible]							
@ V.	—							
AUX.REL.	—							
TRIP	✓							
DATE	12-21-86							
BY	EM / GJW							

LOC. STA 12. BUS 23 BKR 2301 EQUIP. CONTROL ROD DRIVE FD PP 2A IAC, P[illegible]

LOCATION LOAD SIDE OF BKR.

50 / 5	50 / 5		10 & 10	:1
C.T.			SET	:1
			RATIO	:1

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3x 17
5x 11

3x = 17 sec
5x = 11 sec

* NEW RELAYS INSTALLED IN C PHASE AND NEUTRAL; OLD RELAYS SENT TO WYLE LAB. FOR TESTING.

TEST DATA	AΦ	CΦ	±	AΦ	CΦ	AΦ	CΦ	AΦ	CΦ	*CΦ	AΦ	AΦ	CΦ
T.A.	50A	50A	0.5A	50	50	50	50	50	50	50	50	50	50
	5.0A.	5.0A		5.0	5.0	5.0	5.0	5	5	5.0	5.0	50	50
	3.9	3.6				39	3.6	3.9	3.6	3.7	3.9	3.8	3.4
T.I.A.	5.0A	50A		50	5.0	5.0	5.0	50	50	5.0	5.0	5.0	5.0
3X	3x	3x		3X	3X	3X		3X		3X		3X	
17	16.3"	16.4"		17.0	16.2	16.8	16.73	17.4	17.2	17.0	16.9	17.0	17.0
TAP 5X	5X	5X		5X	5X	5X		5X		5X		5X	
11"	10.9"	10.7"		11.35	11.15	11.16"	10.94	11.43	11.25	11.1	11.2	11.2	11.3
				±		±		±		±		=	
				0.5A		0.5a		.5a		0.5A		0.5A	
C.T. MAG.						AΦ		258834		258824		051405	
						CΦ CTR		254343		276129		175405	
	—	—	—			CT		MAG:		LMP		w/	
TRIP	OK	OK	OK	✓ ✓ ✓				ok		OK		✓	
DATE	1-2-75		1-2-75	11-8-77		4-16-79		1/28/81		12-22-83		11-14-84	
BY	GKS-LRT		GKS-LRT					LF		DMR		FAM	

2301

STA. 12 BUS 23 EQUIP. CONTROL ROD DR. FD. PUMP 2A. IAC, PJC RELAYS

REMARKS:

RELAY	MODEL OR STYLE	RANGE
IAV 66	12IAC66B3A	20-80 A (inst) 2.5-5.0 (time)
PJC 11	12PJC11AV1A	.5-2.0 A

AUX.RELAY TRIPPING Service H_2O Pp 3B BKR.

LOAD RES. ALARM Cont. Rm Pnl. 923-1 (w17)

12E-3391

SETTINGS	AØ	CØ	I				
INST.RELAY A	50	50					
INST.LINE A	2000	2000					
TAP A.	5	5	.5				
PRI.A.	200	200	.5				
T.L.	3	3					
[illegible]	[illegible]						
[illegible]TE	[illegible]		[illegible]				
[illegible]	[illegible]		[illegible]				

COPY — CURRENT & VOLTAGE RELAYS

OPER. TYPE LOCATION *Line Side of Bkr.*

RATIO *CT φ 200/5 ⏚ 50/5* SET *[illegible] 10* :1

AUX.C.T. LOC. RATIO SET :1

POL. TYPE LOC. RATIO :1

				As Found	*As Left*			
TEST DATA	*A φ C φ*	*⏚*	*A φ C φ ⏚*	*A φ C φ ⏚*	*A φ C φ ⏚*			
INST.A.	*50 50*	*[illegible]*	*[illegible] 0.5*	*50 50 0.5*	*[illegible] 0.5*			
TAP	*5.0 5.0*	*0.5*	*5.0 5.0 0.5*	*5.0 5.0 .5*	*5.0 5.0 0.5*			
T.L.	*3.5 3.3*		*3.5 3.3*	*3.5 3.3*	*3.4 3.4*			
MIN.A.	*5.0 5.0*		*5.0 5.0*	*4.8 5.0*	*5.0 5.0*			
% TAP *15*	*[illegible]*		*[illegible]*	*[illegible]*				
SEC. *17"*	*17.0 16.5*		*17.0 16.9*	*17.3 [illegible]*	*[illegible]*			
% TAP *25*	*[illegible]*		*[illegible]*	*[illegible]*				
SEC. *11"*	*11.3 [illegible]*		*11.3 [illegible]*	*[illegible]*	*[illegible]*			
DIR.SEN.								
Ammeter			*[illegible]*	*[illegible]*	*[illegible]*			
Timer			*[illegible]*	*[illegible]*	*[illegible]*			
C.T.MAG.	*[illegible]*	*[illegible]*	*w/[illegible]*	*—*	*[illegible]*			
@ V.								
AUX.REL.								
TRIP	*OK*	*OK*	*✓*	*✓*	*✓*			
DATE	*1-15-9?*	*[illegible]*	*[illegible]*	*[illegible]*	*[illegible]*			
	[illegible]	*[illegible]*	*[illegible]*	*[illegible]*	*[illegible]*			

3407

LOC. *STA 12 Bus 34* EQUIP. *Service H - Pr 3B IAC, PTC* RELAYS

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DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4-20-88

CE CO./AE PERSON QUESTIONED: JF KELLY / C M LAUNI

COGNIZANT NRC PERSON K Parkinson

LOCATION/AREA WHERE QUESTION WAS ASKED: Admin Auditorium

QUESTION/ITEM DISCUSSED: Mr Parkinson asked to review all testable check valves that constitute a high/low pressure interface for the 87-50 concerns.

FOLLOW-UP ACTIONS:

RESOLUTION: Parkinson was shown the P&ID's for these valves and that the bypasses around them had ~~[illegible]~~ a normally closed manually operated valve in the series. He found no problem with the Dresden valve arrangement.

PERSONNEL INVOLVED IN RESOLUTION: C. M Launi
J F Kelly

FOLLOW-UP ACTION IF ANY:

RR30

DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4-19-88

CE CO./AE PERSON QUESTIONED: C. Ruth

COGNIZANT NRC PERSON K. Parkinson

LOCATION/AREA WHERE QUESTION WAS ASKED: NRC area (auditorium)

QUESTION/ITEM DISCUSSED: Breaker Coordination for Common Bus Concern: Procedures address removal of non-safe shutdown loads from safe shutdown AC buses to defeat spurious operation and coordination issues. There is no such procedure for the 125Vdc. How do we know 125Vdc will be available?

FOLLOW-UP ACTIONS:

RESOLUTION: For 125Vdc control circuits (breaker controls) the 125V dc is not really necessary because the AC breakers can be manually positioned. Procedures already cover this. Electrical Control will be used if available; if not manual configuration will apply. For the diesel generators, DC is used for field flash. Although the generator is likely to come up on residual flux, 125V dc should be available to flash it if necessary. The study that was initiated in response to Quad Cities' concerns is still in progress and will clarify our position on the 125V dc coordination.

PERSONNEL INVOLVED IN RESOLUTION: C. Ruth, F. Fischer, R Whalen, B. Barth

FOLLOW-UP ACTION IF ANY: Complete the DC coordination study

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DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4/20/88

CE CO./AE PERSON QUESTIONED: B Barth

COGNIZANT NRC PERSON K Parkensen

LOCATION/AREA WHERE
QUESTION WAS ASKED: Auditorium

QUESTION/ITEM
DISCUSSED: routings cable tabs - conduit routings cables 76152, 76153 battery chargers pos + neg polarity cables

FOLLOW-UP
ACTIONS: NONE

RESOLUTION: EXPLAINED THAT 250V DC BATT CHARGERS ARE NOT REQUIRED FOR SSD. ACCEPTED BY MR. PARKINSON

PERSONNEL INVOLVED IN
RESOLUTION: C. E. RUTH

FOLLOW-UP ACTION IF ANY: NONE. COMPLETED.

Revision 8
April 1992

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DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4/20/88

CE CO./AE PERSON QUESTIONED: B Barth

COGNIZANT NRC PERSON J Holmes

LOCATION/AREA WHERE QUESTION WAS ASKED: Cribhouse

QUESTION/ITEM DISCUSSED: what function does device have which is located on side of SW pumps? oil trip water measurement

FOLLOW-UP ACTIONS: device on side of SW pump is oil cooler, cable measures lowlock level, not required for emergency operation.

RESOLUTION:

PERSONNEL INVOLVED IN RESOLUTION: B Barth
R Black

FOLLOW-UP ACTION IF ANY:

RR30

DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: ______________

CE CO./AE PERSON QUESTIONED: B Bach

COGNIZANT NRC PERSON J Holmes

LOCATION/AREA WHERE QUESTION WAS ASKED: Cribhouse

QUESTION/ITEM DISCUSSED: Is an exemption request needed for detection in cribhouse? ~~and~~ Was any submitted?

FOLLOW-UP ACTIONS: ______________

RESOLUTION: An exemption request was not submitted for detection. ~~Only~~ Exemption request were needed for "redundant service water pumps and DG cooling water pumps".

PERSONNEL INVOLVED IN RESOLUTION: C. M Lamm

FOLLOW-UP ACTION IF ANY: NONE

Revision 8
April 1992

DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4/20/88

CE CO./AE PERSON QUESTIONED: B Barth/CKsobiech

COGNIZANT NRC PERSON J Holmes

LOCATION/AREA WHERE QUESTION WAS ASKED: Cribhouse

QUESTION/ITEM DISCUSSED: ability of electrical cables to withstand water spray

FOLLOW-UP ACTIONS:

RESOLUTION: Cables are jacketed with neoprene or thermoplastic materials that are not vulnerable to water.

PERSONNEL INVOLVED IN RESOLUTION: C.E. RUTH

FOLLOW-UP ACTION IF ANY:

RR30

DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4-19-88

CE CO./AE PERSON QUESTIONED: C.E. RUTH / B. M. BARTH

COGNIZANT NRC PERSON K. PARKINSON

LOCATION/AREA WHERE QUESTION WAS ASKED: NRC Area - Auditorium

QUESTION/ITEM DISCUSSED: Potential for spurious operation of shutdown cooling valve 3-1001-2C, which constitutes a high-to-low pressure interface.
Breaker is racked out when valve is not required to be open. Procedures are in place to cover this. It was not originally identified as H/L because the spool piece to fuel pool cooling was to be installed only during an outage. However, due to rad exposure concerns, the spool piece is now left in at all times. Need to address dc-to-dc hot short.

FOLLOW-UP ACTIONS: Route cables for 3-1001-1A and 1B in addition to 2C (addressed separately)

RESOLUTION: Cable routings showed that the inboard valve cables have short routes confined to the east side of the drywell, while the outboard valve cables are routed on the west side of the drywell. There are other power cables of similar voltage in the same trays, but separation ensures that the inboard and outboard valves will not spuriously energize simultaneously.

PERSONNEL INVOLVED IN RESOLUTION:
B. M. BARTH
R. E. BROWN
C. E. RUTH

FOLLOW-UP ACTION IF ANY:

DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4/20/88

CE CO./AE PERSON QUESTIONED: F Fischer; CERuth

COGNIZANT NRC PERSON K Parkensen

LOCATION/AREA WHERE QUESTION WAS ASKED: Auditorium

QUESTION/ITEM DISCUSSED: Want to review station mod review process admin procedure

FOLLOW-UP ACTIONS: PROVIDE DAP 5-1, "PLANT MODIFICATION PROGRAM" ~~THIS CHECKLIST COVERS~~ CHECKLIST 5-1C COVERS FIRE PROTECTION (BARRIERS, SPRAYS, ETC) AND 5-1D COVERS ASSOCIATED CIRCUITS ASPECTS.

RESOLUTION:
PROCEDURES NOTED AND ACCEPTED

PERSONNEL INVOLVED IN RESOLUTION:
R. LEWIS
B. BARTH
C.E. RUTH, R. JOHNSON

FOLLOW-UP ACTION IF ANY:

DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4-20-88

CE CO./AE PERSON QUESTIONED: F.W. FISCHER/ C.E. RUTH

COGNIZANT NRC PERSON KEITH PARKINSON

LOCATION/AREA WHERE
QUESTION WAS ASKED: ADMIN AUD

QUESTION/ITEM
DISCUSSED: REQUESTED A COPY OF FUSE REPLACEMENT EXEMPTION REQUEST FOR THE SWING DG CONTROL CIRCUIT.

FOLLOW-UP
ACTIONS:

RESOLUTION: PROVIDED TO KEITH ON MORNING 04-21-88

PERSONNEL INVOLVED IN
RESOLUTION: C.M. LAUNT

FOLLOW-UP ACTION IF ANY:

DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4-20-88

CE CO./AE PERSON QUESTIONED: C.E. RUTH

COGNIZANT NRC PERSON K. PARKINSON

LOCATION/AREA WHERE QUESTION WAS ASKED: NC area (auditorium)

QUESTION/ITEM DISCUSSED: Cables in common enclosure with service water pump feeds: Keith selected trays 401T, 411T, 411B (in the crib house) from the SWP cable route.

FOLLOW-UP ACTIONS: Provided copies of report S108 pages that apply to those trays (401T, 411T, 411B). Keith then selected some power cables from those trays and asked what protection devices are on those cables.

RESOLUTION: Pertinent cable tabulations, wiring diagrams and key diagrams/schematics showed all cables to have protective devices.

PERSONNEL INVOLVED IN RESOLUTION: C.E. Ruth

F.W. Fischer

FOLLOW-UP ACTION IF ANY:

DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4-20-88

CE CO./AE PERSON QUESTIONED: C.E. RUTH

COGNIZANT NRC PERSON K. PARKINSON

LOCATION/AREA WHERE QUESTION WAS ASKED: NRC area (auditorium)

QUESTION/ITEM DISCUSSED: ERVs: Keith contends that ERVs are a High-to-Low pressure interface. We have documentation that NRR has agreed with our own contention that, since there is no low pressure piping involved, they are not such an interface and therefore multiple spurious signals need not be considered. Keith requested cable routes for all five ERVs so he can determine whether the potential for multiple ERV spurious operations exists. We routed the 3A valve cables in advance.

FOLLOW-UP ACTIONS: Route cables for valves 3B, 3C, 3D, and 3E.

RESOLUTION: Routing is together in tray. Keith says that we should consider that a fire affecting this tray causes all five valves to spuriously open, rapidly blowing down the vessel. However, based on NRR concurrence with our position, we analyzed for time to core exposure for the spurious opening of only one ERV.

PERSONNEL INVOLVED IN RESOLUTION: J. F. Kelly
C. M. Launi

FOLLOW-UP ACTION IF ANY: The audit team will write up this concern as an unresolved item. It will be up to NRR and Region III to decide whether or not the ERVs should be regarded as a high-to-low pressure interface.
IN LETTER TO NRR DATED 7-23-84 NRR WAS INFORMED OF THIS AND AGREED { IN A TELECONFERENCE IN JULY 1987.
CECo's POSITION

Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690 - 0767

July 23, 1987

Mr. Thomas E. Murley, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Quad Cities Station Units 1 and 2
"10 CFR 50, Appendix R Requirements
For High-Low Pressure Interfaces"
NRC Docket Nos. 50-254 and 50-265

Dear Mr. Murley:

In preparation for the 10 CFR 50, Appendix R audit, Commonwealth Edison Company (CECo) is performing a review of the Quad Cities Station's Appendix R Safe Shutdown Analysis. One of the issues raised by this review concerns the applicability of the guidance provided in Generic Letter 86-10 Enclosure 2, Section 5.3.1, regarding the analysis of multiple "hot shorts" in electrical circuits involving high-low pressure interfaces, to the solenoid operated reactor relief valves, 1(2)-203-3A, B, C, D and E, the active components of the Automatic Blowdown System (ABS).

It is the position of CECo that the relief valves do not constitute a high-low pressure interface for the purposes of Appendix R analysis. Thus, analysis of these valves is not subject to the consideration of multiple "hot shorts" in the individual valve control circuitry or ADS circuitry as required by Generic Letter 86-10. The basis for this position is contained in the guidance provided by NRC in Generic Letter 81-12 and its clarification. The Staff's concern with high-low pressure interfaces is that a single fire could cause redundant reactor coolant boundary valves to open, resulting in a fire-initiated LOCA through the subject interface. This concern does not exist in regard to the relief valves for two reasons. First, the relief valves are not redundant coolant system isolation valves. The opening of any individual relief valve will create a flow path for reactor coolant through the valve to the suppression pool located in the pressure-suppression chamber (torus) portion of primary containment. The second reason the valves are not high-low pressure interfaces is this flow of reactor coolant does not constitute a LOCA since the coolant is maintained in a recoverable location (i.e., the torus which is expressly designed for this purpose) within primary containment. Thus, no fire-induced LOCA is possible due to spurious operation of the relief valves and therefore the valves are not considered to be high-low pressure interfaces for the purpose of Appendix R analysis.

The response to Generic Letter 81-12 was provided by CECo for the Quad Cities Station by letter dated July 1, 1982. As stated there, "The only identified high-to-low pressure interface with dual motor operated isolation valves....are located on the Residual Heat Removal System shutdown cooling pump suction lines...." In order to prevent a fire-induced spurious operation from causing a LOCA through this interface, it was proposed that the normally closed RHR shutdown cooling valve be locked in a deenergized position at the appropriate motor control center. The NRC staff reviewed the response and found it acceptable, as documented in the December 30, 1982 Safety Evaluation Report, Section 3.4.3.

In conclusion, it is CECo's position that the consideration of multiple "hot shorts" in electrical circuits involving high-low pressure interfaces to the ADS and related solenoid operated relief valves is outside the requirements of Generic Letters 86-10 and 81-12. We further believe that your staff has accepted this position in the past as evidenced in your December 30, 1982, Safety Evaluation.

Please direct any questions you may have regarding this matter to this office.

Very truly yours,

Irene M. Johnson

I. M. Johnson
Nuclear Licensing Administrator

lm

cc: NRC Regional Administrator
NRC Resident Inspector - Quad Cities
T. Ross - NRR

3387K

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DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4/21/88

CE CO./AE PERSON QUESTIONED: B Barth

COGNIZANT NRC PERSON R Hodor

LOCATION/AREA WHERE QUESTION WAS ASKED: Auditorium

QUESTION/ITEM DISCUSSED: What is tech spec marking requirement?

FOLLOW-UP ACTIONS: Provided information in Tech spec and DAP 7-1

RESOLUTION:

PERSONNEL INVOLVED IN RESOLUTION: B Barth

FOLLOW-UP ACTION IF ANY:

DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4-21-88

CE CO./AE PERSON QUESTIONED: C-E-RUTH

COGNIZANT NRC PERSON K. PARKINSON

LOCATION/AREA WHERE QUESTION WAS ASKED: NRC AREA (AUDITORIUM)

QUESTION/ITEM DISCUSSED: CURRENT TRANSFORMERS:
SAFE SHUTDOWN REPORT SAYS CARBON TRACKING WILL RENDER THEM HARMLESS. THIS REQUIRES JUSTIFICATION.

FOLLOW-UP ACTIONS: PROVIDE COPY OF S&L CALCULATION 6731-EAD-1 AND R.M HIGDON MEMORANDUM TO CER DATED OCTOBER 11, 1985

RESOLUTION: ABOVE CALC REVIEWED AND ACCEPTED BY MR. PARKINSON.

PERSONNEL INVOLVED IN RESOLUTION: C-E-RUTH

FOLLOW-UP ACTION IF ANY:

DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22, 1988

DATE: 4-21-88

CE CO./AE PERSON QUESTIONED: C. E. RUTH

COGNIZANT NRC PERSON K. PARKINSON

LOCATION/AREA WHERE QUESTION WAS ASKED: TURBINE BLDG. CENTRAL ZONE GROUP

QUESTION/ITEM DISCUSSED: WRAPPED TRAYS; WHAT IS IN THERE THAT REQUIRED WRAPPING?

FOLLOW-UP ACTIONS:

RESOLUTION: STATED ~~[illegible]~~ THAT 2/3 DIESEL GENERATOR AUXILIARIES & ASSOCIATED CABLES ARE THERE; NEEDS PROTECTION IN CENTRAL ZONE GROUP (FIRE AREA TB-II); WRAP ENDS IN WESTERN ZONE GROUP (FIRE AREA TB-III).

PERSONNEL INVOLVED IN RESOLUTION: C. E. RUTH

FOLLOW-UP ACTION IF ANY:

DRESDEN STATION APPENDIX "R" AUDIT QUESTIONNAIRE FORM
APRIL 18 - 22,1988

DATE: 4-21-88

CE CO./AE PERSON QUESTIONED: C.E. RUTH

COGNIZANT NRC PERSON K. PARKINSON

LOCATION/AREA WHERE QUESTION WAS ASKED: REACTOR BLDG. - UNIT 2

QUESTION/ITEM DISCUSSED: DIVISION II TRAY JUNCTION WITH DIV. I: MR. PARKINSON OBSERVED JUNCTION OF REDUNDANT DIVISIONS ABOVE BUS 23-1 IN RB 2. HE INQUIRED WHAT SORT OF CABLES WOULD PASS THROUGH THIS JUNCTION.

FOLLOW-UP ACTIONS:

RESOLUTION: STATED THAT CABLE ROUTING/SEGREGATION CRITERIA ALLOW BOP CABLES TO CROSS THIS JUNCTION. FIRE BREAKS ARE INSTALLED NEAR THIS POINT.

PERSONNEL INVOLVED IN RESOLUTION: C-E RUTH

FOLLOW-UP ACTION IF ANY:

Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690

February 1, 1989

Mr. A. Bert Davis
Regional Administrator
U.S. Nuclear Regulatory Commission
Region III
799 Roosevelt Road
Glen Ellyn, IL 60137

Subject: Dresden Nuclear Power Station Units 2 and 3
Response to Notice of Violation
Nos. 50-237/88010-06 and 50-249/88012-06
NRC Docket Nos. 50-237 and 50-249

Reference: J.J. Harrison letter to Cordell Reed dated January 3, 1989 including Notice of Violation concerning improper storage of transient combustible liquids.

Dear Mr. Davis:

The referenced letter provided the results of special safety inspections conducted by Messrs. J. Holmes, R. Hodor, and K. Parkinson on April 18-22, May 11-3, August 15 and December 13, 1988, of Fire Protection activities at Dresden Nuclear Power Station. During the course of these inspections, certain activities appeared to be in noncompliance with NRC requirements. As requested, a response to Item 2 of the Notice of Violation is provided in the Attachment.

Commonwealth Edison understands the significance of the issues identified in the Notice of Violation and has implemented corrective actions to prevent recurrence. Although several other aspects of the Inspection Report warrant clarification, our comments will be provided via a separate transmittal.

If there are any further questions regarding this response, please contact this office.

Very truly yours,

Henry E. Bliss

H. E. Bliss
Nuclear Licensing Manager

lm

Attachment

cc: B.L. Siegel - Project Manager, NRR
S.G. DuPont - Senior Resident Inspector, Dresden

5528K

ATTACHMENT

COMMONWEALTH EDISON COMPANY

REPLY TO NOTICE OF VIOLATION

VIOLATION

10 CFR 50.40(a) requires that each operating nuclear power plant have a fire protection plan that satisfies General Design Criterion 3 of Appendix A to 10 CFR Part 50. It further requires that the plan shall describe specific features necessary to implement the program such as administrative controls and the means to limit fire damage to structures, systems, or components important to safety so that the capability to safely shutdown the plant is ensured.

Section B.2 of the licensee's response to the Guidelines of Appendix A to Branch Technical Position APCSB 9.5-1 as accepted in the 1980 Supplemental Safety Evaluation Report indicates that effective administrative measures will be implemented to prohibit bulk storage of combustible materials inside or adjacent to safety-related buildings or systems during operation or maintenance periods.

Contrary to the above, during a previous inspection conducted on April 12, 1988, an NRC inspector observed twenty 55-gallon drums of lubricating oil stored in a safety-related area on Elevation 517' - 6" (in the southwest corner) of the Unit 2 Reactor Building. This condition existed from March 31 to April 13, 1988. This is a Severity Level IV violation (Supplement I).

BACKGROUND INFORMATION

As part of the Unit 3 refuel outage, the Unit 3 HPCI turbine was inspected and required maintenance performed. This inspection included draining the lube oil reservoir and either cleaning or replacing the oil. The Unit 3 HPCI lube oil reservoir was drained into twenty 55 gallon drums located on Elevation 517 of the Unit 3 Reactor Building (southeast corner). Shortly after this job was completed, other outage work required the "3A" LPCI heat exchanger be accessible. To obtain access to the "3A" LPCI heat exchanger, the oil drums had to be relocated. The Fire Marshal was contacted and determined that the barrels could be moved to Unit 2 Reactor Building Elevation 517 with no additional fire protection, based on:

(1) it being a low traffic area,

(2) the fire detection system was available,

(3) safe shutdown could still be achieved,

(4) the characteristics of the lube oil (being difficult to ignite and having a relatively high flash point),

(5) no other work being performed or combustibles being stored in the area, and

(6) the increase in fire loading did not exceed a low fire loading (i.e., less than 100,000 BTU/ft^2), based on the storage area involved, as defined in the NFPA handbook (15th edition).

Despite the above, CECo understands the inspector's concern regarding the storage of significant quantities of transient combustible liquids in safety related areas of the plant without additional compensatory measures.

CORRECTIVE ACTION AND RESULTS ACHIEVED

On April 13, 1988, the barrels of lubricating oil were removed from the Unit 2 Reactor Building, thus eliminating the concern.

CORRECTIVE ACTION TAKEN TO AVOID FURTHER NON-COMPLIANCE

Dresden Administrative Procedure (DAP) 3-3, "Control of Transient Combustible Storage Areas and No Smoking Areas," will be revised by the Fire Marshal and a Fire Protection Engineer from Support Services. The revision will ensure that routine bulk storage of transient combustibles will not exist in the plant except in designated areas. The revision will establish temporary bulk storage guidelines defining the time a specified amount of transient combustibles will be allowed in a non-designated area and compensatory measures to be taken during that time. In addition, future reviews will be based strictly on the total amount of combustibles allowed in an area. The procedure revision will be completed by March 31, 1989.

DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED

Full compliance was achieved on April 13, 1988

5528K

Tab 8

DRESDEN 2 & 3

FIRE PROTECTION PROGRAM DOCUMENTATION PACKAGE

Inspection Report No. 50-237/88010 and 50-249/88012

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

RECEIVED
JAN 25 1989

JAN 23 1989

Docket No. 50-237
Docket No. 50-249

Commonwealth Edison Company
ATTN: Mr. Cordell Reed
Senior Vice President
Post Office Box 767
Chicago, IL 60690

Gentlemen:

This refers to the special safety inspection conducted by Mr. J. Holmes of this office on October 22-23, 1987; May 5-6, 11, 12 and December 21-23, 1988; and January 20, 1989, of activities at Dresden Nuclear Power Station, Units 2 and 3 authorized by NRC Operating Licenses No. DPR-19 and No. DPR-25 and to the discussion of our findings with Mr. E. D. Eenigenburg at the conclusion of the inspection. This inspection was conducted to review allegations regarding deficiencies in fire wrap installations and the training provided to new installers.

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.

No violations of NRC requirements were identified during the course of this inspection.

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

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

JJHarrison

J. J. Harrison, Chief
Engineering Branch

Enclosure: Inspection Reports
No. 50-237/88030(DRS);
No. 50-249/88031(DRS)

See Attached Distribution

JAN 23 1989

Distrubtion

cc w/enclosure:
H. Bliss, Nuclear
  Licensing Manager
J. Eenigenburg, Plant Manager
DCD/DCB (RIDS)
Licensing Fee Management Branch
Resident Inspector, RIII
Richard Hubbard
J. W. McCaffrey, Chief, Public
  Utilities Division

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-237/88030(DRS); 50-249/88031(DRS)

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

Licensee: Commonwealth Edison Company
Post Office Box 767
Chicago, IL 60690

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

Inspection At: Morris, IL 60450

Inspection Conducted: October 22-23, 1987; May 5-6, 11, 12 and December 21-23, 1988; and January 20, 1989.

Inspector: Jeff Holmes J. Holmes 1/20/89 Date

Approved By: R. N. Gardner, Chief 1/20/89 Date
Plant System Section

Inspection Summary

Inspection on October 22-23, 1987; May 5-6, 11, 12 and December 21-23, 1988; and January 20, 1989 (Report Nos. 50-237/88030(DRS); 50-249/88031(DRS)
Areas Inspected: Special safety inspection into allegations of deficiencies in the fire wrap installations and deficiencies in the training provided to new installers.
Results: No violations or deviations were identified.

- The inspection concluded that while two of the three alleger's concerns were substantiated, no violations of NRC regulatory requirements were identified. With regard to the alleger's third concern, there was no evidence found to support the allegations that there was a lack of independence between Quality Control and Production activities.

DETAILS

1. Persons Contacted

Commonwealth Edison (CECo)

*E. D. Eenigenburg, Station Manager
E. Armstrong, Regulatory Assurance Supervisor
*B. Barth, Technical Staff Engineer
R. Black, Assistant Fire Marshal
*M. Dillon, Fire Marshal
T. G. Hausheer, Fire Protection Engineer, Production Services
*K. Peterman, Regulatory Assurance Supervisor
C. W. Schroeder, Services Superintendent

Transco

G. Jarose, Engineering Manager
L. Anderson, General Foreman
W. Baar, Installer
B. Fatt, Division Quality Assurance Manager
P. Greaney, Installer
B. Leone, Quality Control
D. Marz, Installer
S. Pearson, Quality Control
D. Sisk, Quality Control

U.S. Nuclear Regulatory Commission (U.S. NRC)

S. DuPont, Senior Resident Inspector

*Denotes these person participating in the telecon exit meeting on January 20, 1989.

2. Allegation RIII-87-A-0074

Region III received a telephone call on May 21, 1987, from a former contractor employee at Dresden who contended that deficiencies existed in fire wrap installations and in the training provided to new fire wrap installers. The individual also indicated that there was a lack of independence between Quality Control and Production Activities. Each of the individual's concerns are addressed below:

Concern 1: The training program provided to new installers consisted of requiring the installer to read the procedure and sign a document that indicated that the installers had read and understood the procedure. The training program did not contain any practical demonstrations and new installers were expected to obtain their training on the job.

NRC Review: The allegation was substantiated in that training provided to new installers consisted of having new installers read the procedure and then sign a document showing that the installers had read and understood the procedure. The allegation was also correct in that the

training did not contain any practical demonstration and the new employees were expected to obtain their training on the job.

The Transco procedure for qualification of site craft personnel (PSQAP 2.1) indicates that the indoctrination period varies in length, and scope, and is totally dependent upon the complexity of the functions involved and past experience of the individual. In addition, the procedure indicates that indoctrination is administered either on-the-job or within a classroom environment and is recorded on the "Site Personnel Certification Form" as attestment to qualification by the Transco Field Superintendent.

In discussions with the licensee and Transco, Transco indicated that the individuals who are hired as installers must have a union card which is obtained by apprenticeship with an experienced installer for at least two years. Transco indicated that if the individual installer can follow directions installing insulation, then the individual can follow Transco procedures. Transco indicated that the procedures are required to be read and this takes approximately 15-30 minutes. Afterwards, the Superintendent reviews the procedures with the installers and discusses key points using the specific details and pertinent documents. The installer is then transferred to a Foreman or Leadman. The Foreman or Leadman is responsible for the crew and usually determines the duties of the new installer (the new installer is normally assigned to a member of the crew).

The inspector conducted field walkdowns and reviewed the training records and the installation procedures. The inspector also discussed the Transco training program with several installers, and Quality Control personnel. The Transco employees indicated a mixed opinion regarding the training from excellent to additional training is required. The general consensus was that the General Foreman and Quality Control personnel would insure that an adequate fire wrap was installed.

Conclusion: Based on a detailed review of the field "take-off" records, installation drawings, nonconformance reports, field walkdowns, and interviews with Transco employees, no discrepancies or violations of regulatory requirements were identified. Although the training provided by Transco to new installers may have been weak in certain cases, it appeared that the Transco General Foreman and Quality Control personnel insured that the installation was done according to design criteria.

Concern 2: On-the-job training was given by new employees and therefore untrained new employees were providing on-the-job training to newly hired employees.

NRC Review: This allegation was substantiated. In discussions with Transco and the licensee, they acknowledged that new employees may have been in a position to provide on-the-job training to new employees, but that the General Foreman and Quality Control personnel observed the key parameters in the installation and would have identified an incorrect installation.

Conclusion: Based on detailed review of the field "take-off" records, installation drawings, non-conformance reports, field walkdown, and interviews with Trancso employees, no discrepancies or violations of

regulatory requirements were identified. Although on-the-job training may have been given by new employees, it appeared that the Transco General Foreman and Quality Control personnel insured that the installation was done correctly.

Concern 3: There was a lack of independence between Quality Control and Production Activities in that the Production Superintendent (or General Foreman) was contacting the Quality Assurance Manager and complaining that Quality Control was delaying production. Also, the Production Superintendent controlled the company telephone and truck and prevented Quality Control from using the telephone or truck unless permission was granted from the Production Superintendent or General Foreman.

NRC Review: In discussions with the Quality Assurance Manager, the Manager indicated that telephone calls were received from the field superintendent (or General Foreman) regarding design and installation of the Fire Wrap. The Quality Assurance Manager further indicated that no calls were received regarding Quality Control Inspectors or Quality Control Managers delaying Production. Also, the Quality Assurance Manager indicated that during the exit interviews of the Quality Control Inspectors and Quality Control Managers, no safety issues or issues regarding Production Superintendents contacting the Quality Assurance Manager was discussed.

In addition, the Quality Assurance Manager indicated that Quality Control Inspectors and Quality Control Managers were allowed to use the office telephone for business and not for personal reasons. The Quality Assurance Manager also indicated that the Transco truck was strictly used to transport material and pick-up mail and that permission from the Production Superintendent was required to utilize the company truck.

In discussions with Transco management personnel, Transco indicated that the Quality Control Group was under the direction of the Quality Assurance organization which reported directly to the President of the company and that if any disagreement between production and Quality Control personnel did occur and could not be resolved thru the management organization then it would be resolved by the President of the company.

Conclusion: Based on discussions with the Quality Assurance Manager there was no evidence that the production superintendent (or General Foreman) was contacting the Quality Assurance Manager to report a Quality Control Inspector or Quality Control Manager for delaying production.

In addition, based on discussions with Transco management personnel, the telephone was available for Quality Control, however, the company truck (which was used to transport material) was not available to the Quality Control Group unless permission was granted from the Production Superintendent. The company truck was considered part of the equipment utilized by production and it is not considered unreasonable that the Quality Control Group requested permission to use the company truck.

Based on the above, there was no indication that a lack of independence existed between the Quality Control and Production Activities.

Unit 2 Trackway Fire Wrap Details

The licensee has fire wrapped risers on elevation 517' and 534' consisting of cable tray risers R379 and R380 which interconnect two large sheet metal pull boxes. Transco developed a fire wrap access cover to these pull boxes by using criteria from Transco Detail J6 and Special Drawing EJ 44 (dated January 30, 1987). Due to the numerous physical configurations that may be encountered in the field, 3M allows variances in its application of the material as long as it meets its design criteria. The observed access cover developed by Transco for the licensee appeared to meet the critical criteria such as number of layers, bands, caulking, etc., however, due to its unique design, it was requested that 3M review the installation of this design to ensure that its unique design had not invalidated its fire rating. This is considered an Open Item (237/88030-01(DRS); 249/88031-01(DRS)) pending review of the 3M response.

4. Crib House

During an inspector walkdown, it was observed that a small portion of the fire wrap installation on a junction box did not contain caulk. After the licensee was informed of this concern, the fire wrap was declared - partially degraded.

In discussion with the licensee, the licensee indicated that work had been performed on the junction box and the original fire wrap removed. After work was completed, the wrap was replaced and the caulk not replaced in the lefthand corner of the barrier. The licensee indicated to the inspector that 3M will be conducting training sessions for the installation of the fire wrap for workers and Quality Control personnel at the end of January 1989. The licensee also indicated that the small opening will be recaulked by the end of January 1989.

5. Open Items

Open items are matters that have been discussed with the licensee, which will be reviewed further by the inspector, and which involve some action on the part of the NRC or licensee or both. An open item disclosed during this inspection is discussed in Paragraph 3.

6. Exit Interview

The inspector conducted a telecon meeting with licensee representatives at the conclusion of the inspection and summarized the scope and findings of the inspection. The licensee acknowledged the inspector's comments. The inspector also discussed the likely informational content of the inspection report with regard to documents or processes reviewed during the inspection. The licensee did not identify any such documents or processes as proprietary.

Commonwealth Edison
Dresden Nuclear Power Station
R.R. #1
Morris, Illinois 60450
Telephone 815/942-2920

April 14, 1989

EDE LTR: #89-311

Mr. Ronald J. Israelson
3M Ceramic Materials Department
Building 207-1SC-12, 3M Center
St. Paul, MN 55144-1000

Subject: Review of Installed E-50 Fire Wrap Removable Covers

Dear Mr. Israelson:

As part of the E-50 fire wrap systems installed at Dresden Station during 1987, several configurations were installed that did not follow a standard 3M detail. Deviations from details are permitted as according to 3M drawing 5300-QA, provided that critical design requirements are met. Several of the deviations included installation of removable covers on electrical junction boxes and pull boxes (see attachments). During the installation process, 3M representatives assisted the installers in proper installation procedures and techniques, though few standard detail drawings were formally prepared. The Nuclear Regulatory Commission (NRC), during a review of Dresden Station's 3M fire wrap installations, questioned the practice of installing non-standard designs without the development of special "site-specific" details. The NRC requested that CECo have the installed designs reviewed by 3M Corporation to ensure adherence to E-50 fire wrap system requirements. The attachments list the standard details which are believed to have been followed during the design and installation of the removable covers.

During a 3M E-50 system training session held at Dresden Station on January 25, 1989, you were questioned by the Technical Staff Fire Protection System Engineer regarding the installed configuration of removable fire wrap covers at Dresden Station. At that time, you indicated that the design appeared to meet E-50 fire wrap system critical design requirements.

Dresden Station is requesting 3M Corporation to review the as-built sketches for compliance to E-50 fire wrap system requirements. The Station understands that 3M will provide technical support for it's E-50 fire wrap product at no additional cost to the purchaser. If you require additional design information, please contact Eric Skowron, Technical Staff Fire Protection System Engineer at extension 2353.

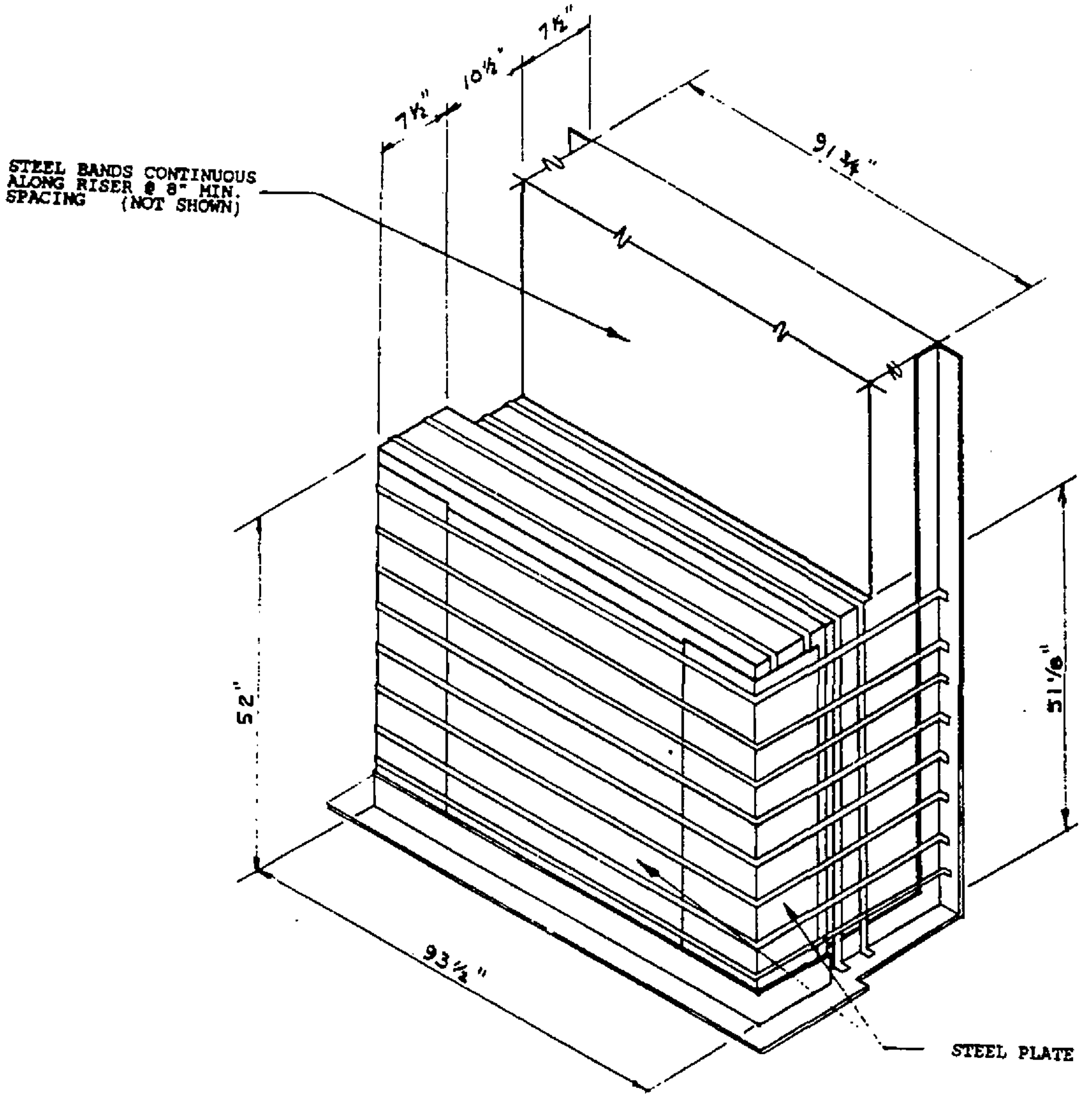
E.D. Eenigenburg
Station Manager
Dresden Nuclear Power Station

EDE:EJS:jmt
Attachments
cc: M. Strait
R. Whalen
M. Dillon
E. Skowron
File/T.S. File (4100)
File/Misc
File/Numerical

3427a

ATTACHMENT 1

Revision 8
April 1992

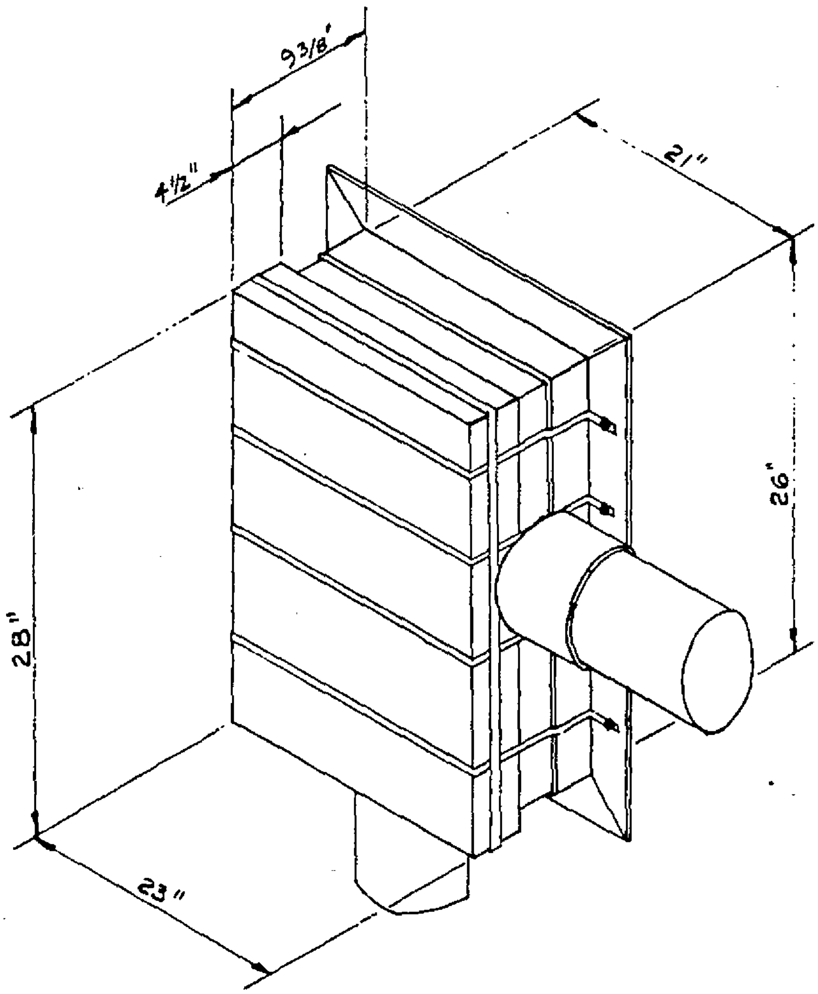


5300-J6	E-50A LAYER REQUIREMENTS
	E-50A 7 CS-195 INTERFACE DETAIL TO WALL
5300-QA	MATERIALS AND GENERAL INSTALLATION REQUIREMENTS:
(P1-8)	E-50 MAT
	CS-195 COMPOSITE SHEET
	CP-25 CAULK
	T-49 ALUMINUM TAPE
	STEEL BANDING
	STEEL COVER (SUBSTITUTED FOR STEEL WOVEN MESH)
6000-EJ59	STEEL BAND ANGLE BRACKET ANCHOR DETAIL
6000-EJ41	STEEL BANDING TRANSVERSESLY CONNECTED TO OTHER BANDS
6000-EJ45	OVERLAP REQUIREMENTS

CABLE TRAY RISERS R379 AND R380

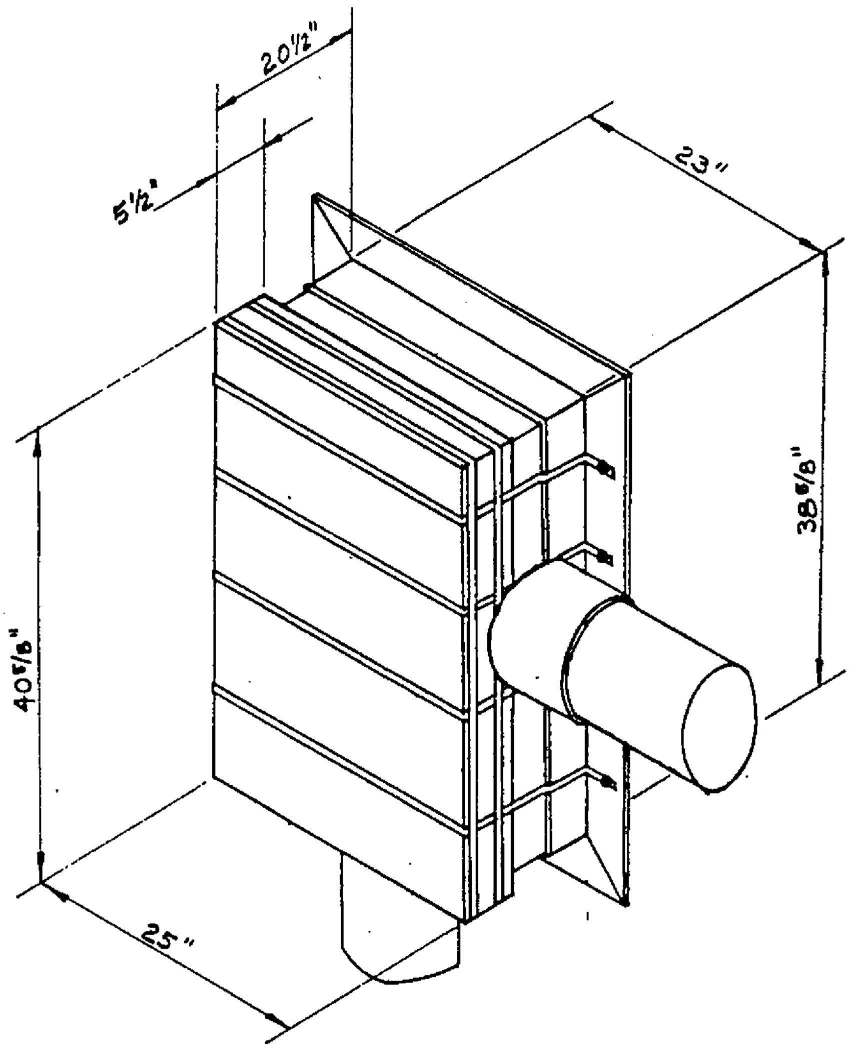
ATTACHMENT 2

Revision 8
April 1992



5300-J6	E-50A LAYER REQUIREMENTS
	E-50A & CS-195 INTERFACE DETAIL TO WALL
5300-QA	MATERIALS AND GENERAL INSTALLATION REQUIREMENTS:
(P1-8)	E-50A MAT
	CS-195 COMPOSITE SHEET
	CP-25 CAULK
	T-49 ALUMINUM TAPE
	STEEL BANDING
6000-EJ41	STEEL BAND ANCHOR DETAIL
6000-EJ45	OVERLAP REQUIREMENTS

JUNCTION BOX 3CB-9



5300-J6	E-50A LAYER REQUIREMENTS
	E-50A & CS-195 INTERFACE DETAIL TO WALL
5300-QA	MATERIALS AND GENERAL INSTALLATION REQUIREMENTS:
(P1-8)	E-50A MAT
	CS-195 COMPOSITE SHEET
	CP-25 CAULK
	T-49 ALUMINUM TAPE
	STEEL BANDING
6000-EJ41	STEEL BAND ANCHOR DETAIL
6000-EJ45	OVERLAP REQUIREMENTS

PANEL 2223-109

3M Center
St Paul, Minnesota 55144-1000
612/733 1110

3M

May 3, 1989

Mr. E.D. Eenigenburg
Station Manager
Dresden Nuclear Power Station
RR #1
Morris, Ill. 60450

Dear Mr. Eenigenburg,

I would like you know that I received your letter dated April 14, 1989. I am familiar with the installations that are detailed on the drawings labelled "Cable Tray Risers R379 and R380", "Junction Box 3CB-9", and "Panel 2223-109". Although I cannot verify that the installations were performed to meet the drawings, I can state that the drawings represent suitable applications of the 3M Fire Protection requirements. Assuming that the installations were performed as described on the drawings, each installation represents a full 1 hour of fire protection.

Please call me at (612)736-3816 if you need any addititonal assistance.

Sincerely,

Ron Israelson

Ronald J. Israelson
3M Technical Service

FPDCC	A-1056
INITIAL	

cc: Eric Skowron - Technical Staff Fire Protection Systems Engineer

Tab 9

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

FEB 28 19[illegible]

Docket No. 50-249

Commonwealth Edison Company
ATTN: Mr. Cordell Reed
Senior Vice President
Post Office Box 767
Chicago, IL 60690

Gentlemen:

This refers to the special safety inspection conducted by Mr. Jeff Holmes of this office on June 4, 1988 through February 8, 1989 of circumstances associated with a fire in the Dresden Nuclear Power Station, Unit 3 Drywell Expansion Gap on June 4, 1988, authorized by NRC Operating License No. DPR-25 and to the discussion of our findings with Mr. E. Eenigenburg at the conclusion of the inspection.

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.

During this inspection, certain of your activities appeared to be in violation of NRC requirements, as described in the enclosed Notice. The inspection showed that actions had been taken to correct the identified violation and to prevent recurrence. Our understanding of your corrective actions is described in Paragraph 2.e. of the enclosed inspection report. Consequently, no reply to the violation is required and we have no further questions regarding this matter at this time.

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

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

Sincerely,

R. W. Cooper

R. W. Cooper, Chief
Engineering Branch

Enclosures:
1. Notice of Violation
2. Inspection Report
No. 50-249/89004(DRS)

See Attached Distribution

Distribution

cc w/enclosures:
H. Bliss, Nuclear
Licensing Manager
J. Eenigenburg, Plant Manager
DCD/DCB (RIDS)
Licensing Fee Management Branch
Resident Inspector, RIII
Richard Hubbard
J. W. McCaffrey, Chief, Public
Utilities Division

NOTICE OF VIOLATION

Commonwealth Edison Company　　　　Docket No. 50-249

As a result of the inspection conducted on June 4, 1988 through February 8, 1989, and in accordance with 10 CFR Part 2, Appendix C - General Statement of Policy and Procedure for NRC Enforcement Actions (1988), the following violation was identified:

Dresden Technical Specification Section 6.2, entitled, "Plant Operating Procedures," requires that detailed Fire Protection Program Procedures be prepared, approved and adhered to.

The licensee's Fire Preventive Procedure 3-2 requires that when cutting or welding activities are in progress care shall be taken not to direct slag from cutting operations through nearby openings.

Contrary to the above, on June 4, 1988, the licensee did not adequately protect penetration X-114, which leads to the combustible drywell liner, from cutting and welding activities. As a result, a fire was initiated in the drywell liner.

This is a Severity Level IV violation (Supplement 1).

The inspection showed that actions had been taken to correct the identified violation and to prevent recurrence. Consequently, no reply to the violation is required and we have no further questions regarding this matter.

FEB 28 1989
Dated

R. W. Cooper II
R. W. Cooper, Chief
Engineering Branch

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-249/89004(DRS)

Docket No. 50-249 License No. DPR-25

Licensee: Commonwealth Edison Company
Post Office Box 767
Chicago, IL 60690

Facility Name: Dresden Nuclear Power Station, Unit 3

Inspection At: Dresden Station, Morris, Illinois

Inspection Conducted: June 4, 1988 through February 8, 1989

Inspector: Jeff Holmes
J. Holmes

2/28/89
Date

Approved By: Ronald N Gardner
Ronald N. Gardner, Chief
Plant Systems Section

2/28/89
Date

Inspection Summary

Inspection on June 4, 1988 through February 8, 1989 (Report No. 50-249/89004(DRS))
Areas Inspected: Announced special safety inspection conducted to review licensee actions with regard to a fire in the Unit 3 Drywell Expansion Gap which occurred on June 4, 1988. This inspection was performed in accordance with NRC Manual Chapter Procedures 64704, and 93702.
Results: Of the areas inspected, one violation was identified in that the licensee failed to adhere to fire prevention procedures (Paragraph 2.e.).

DETAILS

1. Persons Contacted

Commonwealth Edison Company (CECo)

+E. Eenigenburg, Station Manager
+B. Barth, Technical Staff Engineer
+M. Dillon, Fire Marshal
+K. Peterman, Regulatory Assurance Supervisor

U. S. Nuclear Regulatory Commission (NRC)

+S. DuPont, Senior Resident Inspector

+Denotes those participating in the telecon exit meeting on February 8, 1989.

2. June 4, 1988 Drywell Expansion Gap Fire

a. Apparent Origin of the Fire

At 0600 hours on June 4, 1988, with Unit 3 shutdown and in the refueling mode, an air arc cutting activity on the drywell flued head anchor by contractor personnel was in progress in the Unit 3 Reactor Water Cleanup (RWCU) pipeway.

During this time the helper/fire watch observed black smoke, but no flames by the welder's legs near penetration X-114. Initially, the fire watch thought the welder's rubber boots were on fire but after observing that the boots were not on fire, the contractors unhooked their air hoses and climbed up to the adjacent landing to obtain the fire extinguisher.

At approximately 0612, two smoke alarms were received in the control room identifying smoke in the Unit 3 Transversing Incore Probe (TIP) room. At 0615, the control room was notified and at 0619 a second call to the shift engineer was placed.

b. Initial Response and Extinguishment Activities

As previously indicated, the welder climbed to the adjacent landing and obtained a fire extinguisher (which had been placed prior to the start of the welding activity on the drywell flued head anchor). When the welder returned to penetration X-114, he discharged the entire contents of the dry chemical extinguisher into the penetration sleeve.

At approximately 0630, the fire brigade reported that the fire appeared "out," however, the fire brigade personnel discharged a second dry chemical extinguisher into the penetration sleeve.

At approximately 0645, the day shift foreman arrived for duty. At 0705, the day shift foreman, while enroute to the Unit 3 RWCU area (fire scene) noticed a haze on the ground floor of the Unit 3 Reactor Building.

At 0720, an alarm was received at the Center Desk which indicated smoke above Unit 3, Reactor Building, East Accumulator Bank. The day shift foreman returned to the Unit 3 Ground Floor and recognized that the symptoms were similar to those which occurred during the 1986 Unit 3 drywell liner fire. The day shift foreman telephoned the shift engineer and recommended sounding the fire siren, evacuating the Reactor Building and informing the fire brigade to spray water into the penetration sleeve.

c. Extinguishment of the Fire

At 0730, control room personnel sounded the fire alarm and announced over the PA system for personnel to evacuate the Reactor Building. After the arrival of the fire brigade at the fire scene, a walkdown and size-up of the fire was performed. At approximately 0745 the fire brigade applied water to the X-114 penetration. The water was applied to the X-114 penetration until 0800. At 0940 the Fire Marshal, Mechanical Maintenance Foreman and the Rad Chem Technician entered the drywell and determined, by use of a heat gun, that no hot spots existed in the drywell liner. The licensee estimates that approximately 500 gallons of water were used to cool and extinguish the drywell liner fire.

d. Licensee's Followup Actions to the 1988 Drywell Gap Fire

- The station manager issued a welding and cutting stop work order on June 4, 1988 at approximately 0800. The release of the welding and cutting stop work order would be allowed after a Projects and Construction (PACS) walkdown with subsequent Fire Marshal or designee approval prior to work resumption.

- Daily station overview of construction jobs would be provided for the remainder of the outage.

- The licensee photographed and video taped the affected area on June 4, 1988.

- On June 6, 1988, a boroscopic examination of the penetration was performed which revealed no discernable damage but identified debris in the annulus. An attempt was made to remove the debris. A similar examination of the annulus could not be performed due to equipment limitations.

- The fire proof wrapping utilized on penetration X-113 was removed and stored in a quarantined area for further inspection.

- The station manager and the site PACS superintendent conducted meetings on June 6, 1988, to discuss the drywell gap fire event with all craft personnel and to emphasize the need to adhere to the station procedures.

- The drywell sand pocket drains were checked for accumulation of water leakage on June 4, 1988. The licensee indicated only minor dripping was present from the sand pocket drains.

e. <u>Cutting and Welding Procedure</u>

Following the January 1986 drywell gap fire, the licensee upgraded cutting and welding Procedure DAP 3-2 to include the following statement.

> "The exterior steel skin of both the Unit 2 and Unit 3 drywells are covered with a polyurethane foam used during initial construction activities. Although procured as self-extinguishing, the foam has previously been ignited through contact with hot slag from cutting operations on a drywell penetration (see Reference 3). Exercise caution when working around openings that lead to the exterior drywell skin."

The procedure also indicated that when employing a process that generates sparks or slag (cutting, brazing, grinding, etc.) above grating decks, or near floor or wall openings, the deck or opening below the operation shall be covered with suitable noncombustible material. Care shall be taken not to direct the slag stream from the cutting operation through nearby openings.

During the cutting and welding operation that was being performed on June 4, 1988, the contractor did not provide a suitable noncombustible cover for unprotected penetration X-114 which was located only a few feet away. The failure to protect the opening in penetration X-114 during cutting or welding activities was contrary to the licensee's approved fire protection/prevention administrative procedure and is considered a violation (249/89004-01(DRS)).

As part of the licensee's corrective action, the licensee has revised the cutting and welding procedure to require an initial inspection by the station Fire Marshal or designee prior to the start of any cutting and welding activity. This is in addition to the area being inspected by the work group supervisor. The inspector informed the licensee that prior to welding or cutting activities, all drywell penetrations within 35 feet should be packed and then covered with suitable noncombustible material. The inspector also requested that the fire watch inspect the outer covering of the noncombustible materials to ensure that rips, tears and/or openings in the outer covering are repaired should these conditions exist.

f. Evaluation of the 1988 Drywell Gap Fire

On July 20, 1986, with Unit 3 shutdown and defueled, an air arc cutting activity on containment pipe penetration No. X-113 resulted in a fire in the Unit 3 Drywell Expansion Gap. The licensee was requested to address several concerns as presented in NRC Inspection Report No. 50-249/86006(DRS). During this inspection, the licensee was requested to readdress those concerns described in the 1986 NRC Inspection Report. The licensee provided the inspector with the Sargent and Lundy June 4, 1988 Fire Report that indicated the fire occurred in the same location as the January 1986 event and therefore, the evaluation of the effects of the 1986 fire were used as a basis for the current assessment. The report indicated that the 1986 fire burned for a much longer period of time and involved a much wider area and thus the analysis presented in the 1986 report bounded any effects that resulted from the 1988 fire. The report readdressed each of the concerns as presented in the licensee's original response dated May 6, 1986 from D. Farrar, CECo, to J. Keppler, NRC. Based on the premise that NRR accepted the licensee's 1986 response and that this fire was bounded by the original 1986 fire, the licensee has addressed the concerns for the 1988 drywell gap fire.

3. Exit Meeting

On February 8, 1989, a conference call was held with the inspector and the licensee's representatives. The inspector discussed the likely content of this report and the licensee did not indicate that any information discussed during the inspection could be considered proprietary in nature.

Tab 10

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

APR 21 REC'D

Revision 8
April 1992

APR 14 1989

Docket No. 50-237
Docket No. 50-249

Commonwealth Edison Company
ATTN: Mr. Cordell Reed
Senior Vice President
Post Office Box 767
Chicago, IL 60690

Gentlemen:

This refers to the special safety inspection conducted by Mr. J. Holmes of this office on March 16-28, 1989, of activities at Dresden Nuclear Power Station, Units 2 and 3, authorized by NRC Operating Licenses No. DPR-19 and No. DPR-25, and to the discussion of our findings with Mr. E. D. Eenigenburg at the conclusion of the inspection. The inspection was conducted to review allegations regarding unsealed openings inside conduits in fire walls and the use of polyurethane in fire walls.

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.

No violations of NRC requirements were identified during the course of this inspection.

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

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

Sincerely,

R. W. Cooper

R. W. Cooper, II, Chief
Engineering Branch

Enclosure: Inspection Reports
No. 50-237/89008(DRS);
No. 50-249/89009(DRS)

See Attached Distribution

Distribution

cc w/enclosure:
T. Kovach, Nuclear
  Licensing Manager
E. D. Eenigenburg, Plant Manager
DCD/DCB (RIDS)
Licensing Fee Management Branch
Resident Inspector, RIII
Richard Hubbard
J. W. McCaffrey, Chief, Public
  Utilities Division

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-237/89008(DRS); 50-249/89009(DRS)

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

Licensee: Commonwealth Edison Company
Post Office Box 767
Chicago, IL 60690

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

Inspection At: Morris, IL 60450

Inspection Conduced: March 16-28, 1989

Inspector: Jeff Holmes
J. Holmes

April 13, 1989
Date

Approved By: R.N. Gardner
R. N. Gardner, Chief
Plant System Section

4-13-89
Date

Inspection Summary

Inspection on March 16-28 1989 (Reports No. 50-237/89008(DRS); 50-249/89009(DRS))
Areas Inspected: Special safety inspection regarding allegations concerning unsealed openings inside conduits in firewalls and the use of polyurethane in fire walls.
Results: No violations were identified. The inspection concluded that the one allegation was substantiated, however, no violations of NRC regulatory requirements were identified.

DETAILS

1. Persons Contacted

Commonwealth Edison (CECo)

*E. D. Eenigenburg, Station Manager
*K. Deck, Quality Assurance
*M. Dillon, Fire Marshal
*R. Falbo, Regulatory Assurance
*L. Kline, Regulatory Assurance
*D. Roberts, Fire Protection Engineer

Sargent and Lundy (S&L)

*Brian Barth, Technical Staff Engineer

*Denotes those attending March 17, 1989 exit meeting.

2. Allegation RIII-88-A-180

On December 16, 1989, Region III received an allegation that there were unsealed openings inside conduits in the firewalls at the Dresden Nuclear Power Station. In addition, the alleger indicated that pyrocrete masked the presence of polyurethane in the firewalls. Each of the individual concerns are addressed below:

Concern 1: The firewalls at Dresden contain unsealed openings inside conduit penetrations. This allegation was general for all firewalls and no specific areas were received from the alleger.

Does not agree with mark-up on att'd form. JAS

NRC Review: The requirement for sealing conduits which penetrate firewalls is contained in the licensee's updated Fire Hazards Analysis, Section 5.0, entitled "Guidelines of Appendix A to APCSB 9.5-1". This document indicates that conduit and piping should be sealed or closed to provide a fire resistance rating at least equal to that of the barrier. In discussions with the cognizant NRR reviewer on March 28, 1989, the inspector determined that the document only required the licensee to install seals between firewalls and conduits which penetrate the firewall.

The inspector discussed this matter with licensee personnel including the Fire Marshal. The licensee was aware of the conduit seal requirements and indicated that seals had been installed between firewalls and all conduits at the points where the conduits enter or exit the firewalls.

During this inspection, the inspector reviewed a sample of the licensee's completed surveillances of conduits which

penetrate firewalls. These surveillances did not identify any instances of improper conduit seal installations and were determined to be acceptable.

The inspector also selected several representative firewalls for walkdown to determine whether the licensee was complying with the fire seal requirements. During the walkdown, the inspector determined that all required fire seals were installed.

Conclusion: This allegation concerned a perceived need to install seals inside conduit openings for all conduits which penetrate firewalls at the Dresden Station. However, since the licensee was not required to seal these conduit openings and since the inspector determined that the licensee was installing all required fire seals, this allegation was not substantiated.

Concern 2: Pyrocrete covers polyurethane in firewalls.

NRC Review: The licensee's Fire Protection Program includes the Guidelines of Appendix A to APCSB 9.5-1. This document requires the licensee to provide 3 hour rated floors, walls, and ceilings enclosing the separate fire areas identified in the Safe Shutdown Analysis. Deviations in the fire barriers were justified in Exemption Requests and have been reviewed and accepted as identified in the NRC Safety Evaluation Report dated January 5, 1989. Based on review of the pertinent documents, the inspector determined that the licensee was required to remove the polyurethane from the fire walls or demonstrate that the polyurethane in the firewall did not affect the 3 hour rating of the fire barrier.

During this inspection, the licensee indicated that polyurethane was commonly installed in firewalls in the past to prevent air leaks. The plant had previously realized the potential hazard of utilizing polyurethane in firewalls and had hired outside contractors to remove the polyurethane from the firewalls. The licensee indicated to the inspector that the majority of the polyurethane had been removed. However, the licensee indicated that polyurethane covered by pyrocrete remained in a firewall between the turbine building and Unit 2 on elevation 545'-6" at coordinates H and 43 through 44. The licensee had elected to cover the polyurethane with pyrocrete due to high radiation exposure and the possibility of breaching secondary containment.

The licensee also indicated to the inspector that polyurethane without a pyrocrete covering was located

around a 12 inch pipe penetration located between the Units 2 and 3 reactor building on elevation 545'-6" at coordinates 44 and H through J. The licensee indicated that due to radiation concerns the polyurethane had not yet been removed.

For both instances of installed polyurethane, the licensee was in the process of assessing the need to remove the installed polyurethane. The licensee indicated that the assessment will be completed by May 1, 1989.

The licensee also indicated to the inspector that an outside fire protection engineering firm has conducted two fire barrier surveillances which did not identify other instances of installed polyurethane.

Conclusion: This allegation was substantiated in that pyrocrete does cover polyurethane installed in one plant location and polyurethane without a pyrocrete covering exists in another location. However, prior to the allegation the licensee removed and replaced the majority of the polyurethane with an appropriate fire rated barrier or seal. Where the licensee was unable to remove the polyurethane due to high radiation and concerns regarding the breaching of secondary containment, the licensee was performing the required assessment of the effect of the polyurethane on the 3 hour rating of the fire barrier. Therefore, no violations or deviations of NRC requirements were identified.

3. Exit Interview

The inspector met with licensee representatives on March 28, 1989. The inspector discussed the likely content of this report and the licensee did not indicate that any information discussed during the inspection could be considered proprietary in nature.

INSPECTION PLANNER

Revision 8
April 1992

Facility: DRESDEN   Inspection Report No.: 237/89008; 249/89009
Inspection Date(s): MARCH 16/17, 1989   Resident Notified: YES
Inspection Type: Routine ____ Reactive ____ Team ____ Allegation ✓
Other (specify) ____
Announced ✓ Unannounced ____
SALP Weaknesses addressed NO.

Inspection Focus (SALP Weakness, Core Program, Regional Initiative, Event Followup	Module/TI No.	SALP Area/ Rating	Open Item No. (OIL)
FIRE BARRIERS - ALLEGATION	099024B	NR	NO
• UNSEALED CONDUITS			
• PYROCRETE OVER POLYURETHANE			

Plan Prepared By: Jeff Holmes   Date: 3/13/89
Lead Inspector
Plan Approved By: R N Gard   Date: 3/13/89
Section Chief
Plan Reviewed By: [signature]   Date: 3/13/89
Project Section Chief

Attachment 2
Rev. 07/07/88

ALLEGATION ACTION PLAN
ALLEGATION NO. RIII-88-A-180

Licensee: Commonwealth Edison Company

Docket/License No: 50-237 and 50-249/DPR-19 and DPR-25

Assigned Division: DRS

Attached Pertinent Documents: Allegation Documentation

I. Division Action

Allegations regarding unsealed conduit penetrations and the adequacy of of fire resistive material covering polyurethane in fire walls at the Dresden Station were received by Mr. Jeff Holmes on December 16, 1988.

Allegation No. 1

IN CERTAIN SPECIFIC AREAS

The alleger alleged that there are unsealed conduit penetrations thru the fire wall at Dresden. In addition, the alleger alleged that pyrocrete covers the polyurethane in the fire walls.

NRC Action

1. Request the licensee to provide documentation addressing conduit and pyrocrete penetration fire barrier commitments/requirements.

2. Review plant surveillance/maintenance procedures that cover conduit and pyrocrete penetration fire barrier configurations.

3. Interview the plant manager and the fire marshal regarding knowledge of concerns regarding the unsealed conduit and pyrocrete over the polyurethane in fire walls.

4. Conduct an inplant review of ... when specified by Alleger selected installed conduit and pyrocrete fire barrier configurations.

A. Prepared by: Joseph M. Ulie
Joseph M. Ulie
Technical Staff
3-9-89
Date

B. Reviewed by: Ronald N. Gardner
Ronald N. Gardner
Section Chief
3-9-89
Date

C. Approved by: R W Cooper
Richard W. Cooper
Branch Chief
3/9/89
Date

II. Allegation Review Board Action

Allegation Review Board Membership

/___/ Approved As Is

/___/ Approved with Modifications as Documented in Plan.

/___/ Disapproved for Following Reasons:

Allegation Review Board Chairman Date

ALLEGATION/PERIPHERAL ISSUE ACTION PLAN

Concerns and any peripheral issues associated with a concern should be documented on a separate page. Each concern and peripheral issue, if any, should be documented in the followup report as is stated in this plan. If there are several concerns in one area, one page can be used. Otherwise, a separate page should be used for each concern.

/___/ Concern No. ____ /___/ Peripheral Issues Associated with Concern No. ____

I. Action Evaluation: The following method of resolution is recommended (circle):

A. Send to Licensee Requesting Response in _____ Days with RIII Followup*
B. Priority RIII Followup
C. Followup During Routine Inspection Within 60 Days
D. Followup with Assistance from OI
E. No Action - Outside NRC's Charter (describe basis below)
F. No Action - Without Merit (describe basis below)
G. Refer to
H. Other (specify)

* If the proposal is to send to the licensee, the Action Plan should describe the general areas we expect the licensee to address.

II. Inspector's Actions: The following areas at a minimum will be reviewed during the inspection into the above mentioned concern and/or peripheral issue.

A. Objective

B. Methods

1. Persons to be contacted:

a. Plant Manager
b. Station Fire Marshal
c. Other personnel as necessary

2. Documents and/or activities to be reviewed:

3. Time period to be covered:

4. Locations/specific areas to visit:

5. Other areas (specify):

Allegation No. RIII-88-A-180

Tab 11

89013/89012

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

Revision 8
April 1992

JUN 9 1989

JUN 13

Docket No. 50-237
Docket No. 50-249

Commonwealth Edison Company
ATTN: Mr. Cordell Reed
Senior Vice President
Post Office Box 767
Chicago, IL 60690

Gentlemen:

This refers to the routine safety inspection conducted by Mr. J. Holmes of this office on April 3-7 and May 24, 1989, of activities at Dresden Nuclear Power Station, Units 2 and 3, authorized by NRC Operating Licenses No. DPR-19 and No. DPR-25, and to the discussion of our findings with Mr. C. Schroeder at the conclusion of the inspection. The purpose of this inspection was to review the implementation of the licensee's fire protection program.

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.

No violations of NRC requirements were identified during the course of this inspection.

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

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

Sincerely,

R. W. Cooper, Jr.

R. W. Cooper, Chief
Engineering Branch

Enclosure: Inspection Reports
No. 50-237/89013(DRS);
No. 50-249/89012(DRS)

See Attached Distribution

Distribution

cc w/enclosure:
T. Kovach, Nuclear
  Licensing Manager
E. D. Eenigenburg, Station Manager
DCD/DCB (RIDS)
Licensing Fee Management Branch
Resident Inspector, RIII
Richard Hubbard
J. W. McCaffrey, Chief, Public
  Utilities Division
A. Datta, NMSS/IMSB
C. McCracken, NRR/ECEB
A. Krasopoulos, RI/DRS
G. Wiseman, RII/DRS
A. Singh, RIV/DRS
C. Ramsey, RV/DRS

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-237/89013(DRS); 50-249/89012(DRS)

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

Licensee: Commonwealth Edison Company
Post Office Box 767
Chicago, IL 60690

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

Inspection At: Morris, Illinois

Inspection Conducted: April 3-7 and May 24, 1989

Inspector: Jeff Holmes
Jeff Holmes
JUNE 8, 1989
Date

Approved By: Ronald N. Gardner, Chief
Plant System Section
Ronald N. Gardner
6-8-89
Date

Inspection Summary

Inspection on April 3-7 and May 24, 1989 (Reports No. 50-237/89013(DRS); 50-249/89012(DRS))
Areas Inspected: Routine, unannounced safety inspection conducted to review the implementation of the licensee's fire protection program including a followup of licensee action on previous inspection findings. This inspection was conducted in accordance with inspection procedures 64704 and 92701.
Results: Of the areas inspected, no violations were identified. One unresolved item and one open item was identified. The unresolved item concerned the need for the licensee to verify the shelf life of two types of fire fighting foam concentrate utilized for flammable liquid fires (Paragraph 3.e). The open item concerned the need to develop a six month functional test for linear detection (Paragraph 3.a). In addition, the inspector identified and discussed with the licensee the need to develop pre-fire plans for areas such as the main power unit transformer area and hydrogen storage tank areas (Paragraph 3.c) and a need to develop a maintenance program for the hydrogen storage equipment and piping (Paragraph 3.d).

DETAILS

1. Persons Contacted

Commonwealth Edison Company (CECo)

*C. Schroeder, Services Superintendent
*C. Allen, Performance Improvement Supervisor
#D. Barnett, Senior Quality Assurance Supervisor
*R. Black, Assistant Fire Marshal
*K. Deck, Quality Assurance Engineer
*,#M. Dillon, Fire Marshal
*R. Falbo, Regulatory Assurance Assistant
*,#L. Gerner, Production Superintendent
#T. Lewis, Regulatory Assurance
*,#K. Peterman, Regulatory Assurance Supervisor
*D. Roberts, Fire Protection Engineer
*J. Silady, Nuclear Licensing Administrator
*E. Skoron, Technical Staff Engineer
*S. Stiles, Training Supervisor

Sargent and Lundy (S&L)

*Brian Barth, Technical Staff Engineer

Nuclear Regulatory Commission (NRC)

#S. DuPont, Senior Resident Inspector

*Denotes those attending the April 7, 1989 exit meeting.
#Denotes those participating in the May 26, 1989 exit meeting (telecon).

2. Action on Previous Inspection Findings (92701)

a. (Closed) Unresolved Item (237/85033-03(DRS); 249/85029-03(DRS)): The licensee's surveillance procedure (SP 84-6-39) failed to incorporate appropriate test requirements to demonstrate that the sprinkler system was operable in accordance with NFPA 13 in that the procedure did not require flow from the two inch drain valve for wet pipe or dry pipe sprinkler systems.

During the week of January 2, 1989, the licensee provided the inspector with an internal memorandum to E. Eenigenburg from R. Black that stated the following:

"The test flow discharges will be handled through pre-installed drain piping to the station storm drains. On systems where pre-installed drain piping is nonexistent, a 1-1/2 inch maintenance hose will be used. We believe a 1-1/2 inch hose provides adequate flow at our system pressure (125 psi) to ensure water supplies and connections are in order.

The reactor building fire hose risers will be tested by connecting a 1-1/2 inch maintenance hose to the upper most hose station locations and test flowed to station storm drains.

It is our plan to complete the writing and get approvals of the surveillance by April 30, 1989 and perform the surveillance for the first time by June 30, 1989. Additional surveillances will be done on an annual basis."

Based on the licensee's commitment to conduct the drain test, this item is considered closed.

b. (Closed) Open Item (237/85033-06(DRS); 249/85029-06(DRS)): The licensee was requested to provide appropriate acceptance criteria for filling breathing air supply cylinders.

The licensee provided the inspector with the procedure titled "Use of the Cascade Recharging System for Filling Self Contained Breathing Apparatus Bottles," DRP 1310-11, Revision 0. The inspector reviewed the procedure and no discrepancies were identified. Based on the inspector's review of this procedure, this item is considered closed.

c. (Closed) Open Item (237/85033-07(DRS); 249/85029-07(DRS)): The inspectors observed deficiencies on the main carbon dioxide storage tank located on the first floor of the turbine building. The deficiencies included the following:

(1) The access door to the tank compressor motor was missing.

(2) The glass cover to the tanks mercoid switch located inside the access door was missing.

During the Appendix R inspection that ended in December 1988, the inspector had observed that the licensee had taken corrective actions to replace the access door and glass cover to the mercoid switch. Based on the licensee's actions, this item is closed.

d. (Closed) Open Item (237/88010-05(DRS); 249/88012-05(DRS)): The pre-discharge alarm for the diesel room carbon dioxide system is only an audible alarm. The licensee was requested to verify that the audible alarm is sufficient to warn personnel that may be in the area with the diesel operating.

The licensee provided the inspector with an internal document dated January 31, 1989, from R. Black, Assistant Fire Marshal to the Fire Protection File. The document indicates that on December 23, 1988, during the performance of DFPP 4145-1, "Cardox System Semiannual Maintenance Test," which was modified for a running diesel, the audible alarm was heard by the occupants in the room.

Based on the licensee's actions, this item is considered closed.

e. (Closed) Violation (237/88010-06(DRS); 249/88012-06(DRS)): The licensee failed to meet the requirements of their approved fire protection program by permitting the storage of twenty 55-gallon drums of lube oil in a safety-related area.

The licensee provided the inspector with the "Control of Transient Combustibles" procedure that has been revised to include the following:

For medium fire loading of an area (5-25 gallons for flammable liquid, 55-120 gallons for combustible liquid) and high fire loading of an area (25 gallon for flammable liquid, 120-240 gallons for combustible liquid) a transient combustible permit signed by the fire marshal or his designee is required. In addition, compensatory measures are required prior to introducing combustibles into the plant. The fire marshal is also required by the procedure to review the fire hazards analysis for the fire area of concern. The basis for acceptance of a high fire load includes the consideration of equipment and combustibles presently in the area and any suppression or detection systems. Compensatory measures are then established. Based on the review of the updated procedures, this item is considered closed.

3. Routine Fire Protection Program

The Dresden fire protection program utilizes the defense-in-depth concept against hostile fires to ensure that safe shutdown capability is not impaired. The Dresden fire protection program philosophy of defense-in-depth consists of:

a. Fire Prevention
b. Detection and Suppression
c. Mitigation of Fire Damage

The inspector reviewed, on a sample basis, the licensee's administrative procedures and fire protection surveillances. The inspector also walked down several fire protection systems. The results of the inspector's review are as follows:

a. Fire Protection System Surveillances

The licensee's fire protection program requires that the licensee test fire protection equipment and systems that are included in regularly scheduled station operating surveillance procedures. The inspector selected a sample of the licensee's completed surveillance procedures for review. During the review, the inspector determined the following:

(1) Weekly Unit 1 Diesel Fire Pump Operability Surveillance

The licensee's Unit 1 diesel fire pump weekly operability surveillance, DFPP 4123-1, Revision 8, includes the

verification that the fire pump batteries are provided with proper electrolyte level and specific gravity.

In addition, the procedure verifies that the battery charger is operating and that proper oil level is provided in the engine case and right angle gear drive. The inspector reviewed the Unit 1 diesel fire pump weekly operability data sheet dated March 28, 1989, and found the results to be satisfactory.

(2) Quarterly Auxiliary Electric Equipment Room Halon Damper Test

The auxiliary electric equipment room halon damper test, DFPP 4195-3, Revision 2, verifies that dampers required to close prior to discharge are operating as designed.

The auxiliary electric equipment room halon damper test results dated March 14, 1989, were found to be satisfactory.

(3) 18 Month Operating Fire Stop/Break Surveillance

The operating fire stop/break surveillance, DFPP 4175-2, Revision 5, verifies by visual observation that the fire stop/break is intact.

The fire stop/break surveillance dated February 29, 1988, was found to be satisfactory.

(4) Annual Auxiliary Electrical Equipment Room Fire Resistive Structural Steel and Cable Coating Surveillance

The auxiliary electrical equipment room fire resistive structural steel and cable coating surveillance, DFPP 4175-4, Revision 1, requires visual verification of the integrity of the auxiliary electrical equipment room fire resistive structural steel and cable coating. In the surveillance dated January 18, 1989, the licensee identified several areas where structural steel fire proofing was found degraded and initiated a DVR. Based on the licensee's actions to correct the degraded fire proofing, the surveillance was found to be acceptable.

(5) Monthly Fire System Yard Loop Inspection

The fire system yard loop monthly inspection, DAP 11-2, Revision 15, checks equipment such as fire hydrants, hose houses, fire hose reels, fire main valves and other fire equipment.

The fire system yard loop monthly inspection dated March 10, 1989, was reviewed and found to be satisfactory.

(6) Monthly Fire System Inspection For Unit 2

The Unit 2 monthly fire system inspection, DFPP 4114-2, revision 9, visually inspects equipment such as hose reels, fire main valves, fire equipment carts, carbon dioxide systems and other fire equipment.

The Unit 2 monthly inspection results dated March 27, 1989, were reviewed and found to be satisfactory.

(7) Six Month Fire Detection Test

The licensee's smoke detector semiannual maintenance test, DFPP 4185-2, revision 6, verifies the response of the fire detection system.

The inspector requested the last six month channel functional tests conducted for Unit 2 fire zones 1.1.2.1 (elev. 476'-6"), 1.1.2.2 (elev. 517'-6"), 1.1.2.3 (elev. 545'-6"), and Unit 3 fire zones 1.1.1.1 (elev. 476'-6"), 1.1.1.2 (elev. 517'-6"), and 1.1.1.3 (elev. 545-6"). Fire zones 1.1.2.1 and 1.1.1.1 are provided with linear thermal detection. The inspector requested the six month functional test for these areas. However, the licensee had not yet developed a six month channel functional test for the linear thermal detectors in these areas. In discussion with the licensee, the licensee indicated that a recent audit had identified the same concern and that the surveillance was in the process of being developed. The licensee indicated to the inspector that the surveillance will be completed by July 21, 1989. This is considered an open item (237/89013-01(DRS); 249/89012-01(DRS)) pending NRC's review of the surveillance procedure. The inspector reviewed the last six month channel functional test dated January 1989 for Fire Zones 1.1.2.2, 1.1.2.3, 1.1.1.2, and 1.1.1.3. The functional test performed did in some cases identify minor problems. The licensee wrote a work request to address those concerns. Based on review of the surveillance test results and the licensee's actions, the surveillance was found to be acceptable.

b. Personnel Required for Safe Shutdown and Fire Fighting Activities

In the event of a disabling fire which requires evacuation of the Unit 2/3 Common Control Room when both units are operating, it would be necessary to provide sufficient personnel to shutdown the operating reactors and provide manual fire fighting capabilities.

(1) Safe Shutdown Personnel

The licensee has developed alternative safe shutdown procedure EPIP 200-20, titled "Control Room Evacuation/Safe Shutdown," Revision 6, dated February 1989.

The licensee's staff required to implement the alternative safe shutdown procedure requiring the evacuation of the control room is as follows:

Shift Engineer (SE)
Station Control Room Engineer (SCRE)
Unit 2 Shift Foreman (SF)
Unit 3 Shift Foreman (SF)
Engineer Assistant (EA)

Center Desk Nuclear Station Operator (NSO)
Unit 2 Nuclear Station Operator (NSO)
Unit 3 Nuclear Station Operator (NSO)
Unit 1 Level 1 Operator/Equipment attendant
Unit 2 Level 1 Operator/Equipment attendant
Unit 3 Level 1 Operator/Equipment attendant
Utility Level 1 Operator/Equipment attendant

(2) Fire Brigade Personnel

The licensee also has developed "Fire Fighting" procedure EPIP 200-4, Revision 5, dated December 1987, which describes the organization of the fire fighting brigade and delineates the duties of the fire brigade. This procedure indicates that the composition of the fire brigade for all shifts is as follows:

Shift Foreman - Fire Chief
High Voltage Operator - Fire Fighter
Radwaste Roving Operator - Fire Fighter
Unit 2/3 Max Recycle Operator - Fire Fighter
Rover - Fire Fighter

(3) Operations Department Organization

The licensee has developed operations department organization procedure DAP 7-1 which identifies the staffing normally required for operating shifts 1, 2, and 3. The inspector verified that the minimum number of personnel for safe shutdown and fire fighting was included in the procedure.

(4) Conclusion

The inspector requested records to demonstrate that the 12 personnel required to implement the control room evacuation procedure and the 5 personnel required for fire fighting activities were available for three shifts on April 13, 1989, and April 26, 1989. The inspector was provided with copies of the appropriate sections of the shift's engineers and center desk books.

In cases where names were inadvertently left out of the logs, the licensee provided backup documentation to demonstrate that these personnel were available.

The inspector verified, based on the licensee's documentation provided, that the appropriate personnel for the 12 positions were available to implement the control room evacuation safe shutdown procedure. The inspector also verified that in addition to the staff required for safe shutdown, the licensee provided a 5 member fire brigade consisting of a fire chief and four fire fighters. Based on the inspector's review of the licensee's documentation, the inspector determined that on April 13 and 26, 1989 (all shifts), the licensee provided sufficient personnel for safely shutting down the reactors and to support any required fire fighting activities.

c. Pre-Fire Plans

The licensee has developed pre-fire plans for fire in safety-related areas as described in the fire hazard analysis. The pre-fire plans indicated important parameters for each fire area such as access, hazards, fire protection equipment, ventilation, communications, exposures (safety-related equipment), construction, guidelines for attack, etc. In addition, the licensee has provided a schematic for each fire area which also indicates location of fire fighting equipment, communication, access points, etc. It appears that the licensee has developed good fire pre-plans for fighting fires in safety-related areas within the plant. However, the inspector identified that pre-fire plans did not exist for areas such as the hydrogen storage area and main power transformers for Unit 2 and Unit 3. Both of these are non safety-related areas.

The licensee was informed that it would be prudent to develop pre-fire plans for all areas with high combustible loading and/or where special precautions may be required to prevent injury to fire fighting personnel or damage to the plant. The licensee acknowledged the inspector's concern and indicated that plant areas not addressed in the fire hazard analysis such as the main power unit transformers and hydrogen storage areas will be reviewed and pre-fire plans developed by December 31, 1989.

d. Hydrogen Storage

The tank farm and the hydrogen injection storage are two areas at the Dresden site that currently store hydrogen for normal plant operation to provide hydrogen cooling for the turbine generator and also to prevent intergranular stress corrosion cracking in primary piping and equipment. Both of these hydrogen systems are non safety-related.

(1) Tank Farm

According to the licensee, the tank farm was installed in 1968 and is provided with fifty fixed storage vessels capable

of storing 35,000 standard cubic feet (scf) of hydrogen at 1250 pounds per square inch (psi). The extra heavy red brass piping from the tank farm to the regulator is pressurized to approximately 1250 psi. After the hydrogen is stepped down by the regulators to a line pressure of 70 psi, the hydrogen then enters into 150 carbon steel pipe.

The underground pipe for this system is provided with cathodic protection. The system has been designed for automatic operation and is provided with a high flow supply line trip. This hydrogen system has also been provided with alarms such as gas purity, gas pressure, high flow, low flow, low main bank pressure and hydrogen storage reserve bank low pressure.

The inspector toured the hydrogen tank farm and observed that the piping from the relief valves was rusty and was not provided with plastic caps.

(2) Hydrogen Injection Storage

There are two hydrogen supply trailers, each with a total capacity of 125,000 scf. This system has been designed for automatic operation and is provided with trips resulting from reactor scram, low feedwater, low offgas, hydrogen high area alarm, local panel shutdown switch and control room shutdown switch. The piping installed from the hydrogen storage trailers to the plant is 304 stainless steel pipe and is provided with cathodic protection.

The pipe installed inside the building is 300 carbon steel pipe. The inspector toured the hydrogen storage trailers and noted that the pressure regulator cabinet that steps down the pressure from the trailer tanks to the system piping was not securely mounted or protected from trucks that deliver hydrogen. In addition, the inspector observed that the trailers were not provided with chocks to secure the wheels to prevent movement.

The licensee acknowledged the inspector's concerns and indicated that the pressure regulator cabinet would be secured, barrier protection for the pressure regulation cabinet would be installed and that chocks would be provided for the wheels to prevent movement.

(3) Conclusion

In discussions with the licensee, it was identified that no regular maintenance had been performed on the hydrogen tank farm since it was installed in 1968. The licensee indicated that there is no regular inspection or maintenance for pressure regulators, relief valves, interlocks, etc.

For the hydrogen injection storage area which is a relatively new addition, the licensee also indicated that no periodic inspection or maintenance has been established for interlocks,

pressure regulators, relief valves, etc.

The licensee concurred that an evaluation should be performed for the hydrogen tank farm and hydrogen injection system to develop an appropriate maintenance program.

e. Plant Tour

The inspector toured several areas of the Unit 2 and Unit 3 reactor building and turbine building. During this tour, the inspector visually observed several hose stations, extinguishers, sprinkler valves, carbon dioxide valves, emergency lights, and housekeeping. The inspector observed that the equipment was in an apparently well maintained condition. Housekeeping, in general, was good. The inspector informed the licensee that the placement of reflective tags identifying appropriate switches for Appendix "R" safe shutdown equipment (for example, at the 250 vdc bus) would be beneficial to the operator. The licensee indicated that the station is currently assessing the use of reflective tags for identifying appropriate switches for Appendix "R" safe shutdown equipment.

The inspector also indicated to the licensee that the overall outside housekeeping needed to be improved. The licensee concurred with the inspector and indicated that housekeeping will be improved in conjunction with the decontamination efforts. During the tour, the inspector also observed the 750 gallons per minute deluge gun (located in the 2/3 cribhouse) which may be used to assist in fighting a main power unit transformer or hydrogen tank fire. The deluge gun (monitor nozzle) is provided with mechanical gears which allow the operator to change the nozzle elevation. The inspector identified that it was difficult to change the elevation of the deluge gun. The inspector suggested that the deluge gun be included in a preventive maintenance program. The licensee concurred with the inspector and indicated that the two deluge guns at the plant would be disassembled and inspected. After the results of the inspection are known, long term continuing maintenance will be established. The licensee indicated that the inspection for the deluge gun would be completed by April 30, 1989. Also during the tour, the inspector observed that the licensee stored Rockwood 6% foam concentrate (1981) and Ansul AFFF 3% foam concentrate (1981) in five gallon cans in the waste water treatment facility. The licensee maintains approximately 50 gallons of foam concentrate at the waste water treatment facility. The inspector questioned the licensee regarding the shelf life of the foam concentrates. The licensee was not aware of the shelf life of the foam concentrate and the licensee indicated that the foam

concentrate is not sent out for testing to determine if it will perform as intended. The licensee indicated that fire fighting foam concentrate shelf life will be verified and if testing is required, it will be scheduled, or the foam concentrates will be replaced by May 31, 1989. The shelf life of the foam concentrate is considered an unresolved item (237/89013-02(DRS); 249/89012-02(DRS)) pending review of the licensee's actions.

The inspector informed the licensee that it would be prudent to use one type of foam concentrate and that the foam concentrate should be rotated such that the older foam concentrate, if needed, can be used during fire fighting training. The licensee acknowledged and concurred with the inspector's comments.

4. Open Items

Open items are matters that have been discussed with the licensee, that will be reviewed further by the inspector, and that involve some action on the part of the NRC, the licensee, or both. Open items disclosed during the inspection are discussed in Paragraph 3.a.

5. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, or items of noncompliance or deviations. An unresolved item disclosed during the inspection is discussed in Paragraph 3.e.

6. Exit Meeting

The inspector met with the licensee representative on April 7, 1989 and also held a conference call with the licensee on May 26, 1989. The inspector discussed the likely content of this report and the licensee did not indicate that any information discussed during the inspection could be proprietary in nature.

Tab 12

89002/89017/89016

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

ORIG-FILE
cc:
cc:
cc:
cc: K. PETERMAN / R. GALLOO
REVIEW A COMMITMENTS TRACK NC

Revision 8
April 1992

AUG 4

JUL 31 1989

Docket No. 50-010
Docket No. 50-237
Docket No. 50-249

Commonwealth Edison Company
ATTN: Mr. Cordell Reed
Senior Vice President
Post Office Box 767
Chicago, IL 60690

Gentlemen:

This refers to the routine safety inspection conducted by S. G. Du Pont, K. R. Ridgway, D. E. Hills and D. E. Miller, of this office on May 30 through July 14, 1989, of activities at Dresden Nuclear Power Station, Units 1, 2 and 3 authorized by Operating Licenses No. DPR-02, No. DPR-19 and No. DPR-25 and to the discussion of our findings with Mr. E. Eenigenburg and others at the conclusion of the inspection.

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.

During this inspection, certain of your activities appeared to be in violation of NRC requirements, as described in the enclosed Notice. The inspection showed that action had been taken to correct the identified violation and to prevent recurrence. Our understanding of your corrective actions are described in Paragraph 11.b of the enclosed report. Consequently, no reply to the violation is required and we have no further questions regarding this matter at this time.

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,

WDShafer
W. D. Shafer, Chief
Reactor Projects Branch 1

Enclosures:
1. Notice of Violation
2. Inspection Reports
   No. 50-010/89002(DRP);
   No. 50-237/89017(DRP) and
   No. 50-249/89016(DRP)

cc w/enclosures:
T. Kovach, Nuclear
  Licensing Manager
E. D. Eenigenburg, Station Manager
DCD/DCB (RIDS)
Licensing Fee Management Branch
Resident Inspector, RIII
Richard Hubbard
J. W. McCaffrey, Chief, Public
  Utilities Division

NOTICE OF VIOLATION

As a result of the inspection conducted on May 30, 1989 through July 14, 1989, and in accordance with the General Policy and Procedures for NRC Enforcement Actions, (10 CFR Part 2, Appendix C), the following violation was identified:

> Dresden Technical Specification 6.2.A states that detailed written procedures covering preventative and corrective maintenance operations, which could have an effect on the safety of the facility . . . and testing and surveillance requirements shall be prepared, approved and adhered to.
>
> Contrary to the above, ventilation hatches in the Unit 2 drywell left in an improper closed position resulting in excessive upper elevation temperatures during Cycle 11 were due to inadequate maintenance and surveillance procedures.
>
> This is a Severity Level IV violation (Supplement I).

The inspection showed that action had been taken to correct the identified violation and to prevent occurrence. Consequently, no reply to the violation is required and we have no further questions regarding this matter.

7/31/89
Dated

W D Shafer
W. D. Shafer, Chief
Reactor Projects Branch 1

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-010/89002(DRP); 50-237/89017(DRP); 50-249/89016(DRP)

Docket Nos. 50-010; 50-237; 50-249 Licenses No. DPR-02; DPR-19; DPR-25

Licensee: Commonwealth Edison Company
P. O. Box 767
Chicago, IL 60690

Facility Name: Dresden Nuclear Power Station, Units 1, 2 and 3

Inspection At: Dresden Site, Morris, Illinois

Inspection Conducted: May 30 through July 14, 1989

Inspectors: S. G. Du Pont

K. R. Ridgway

D. E. Hills

D. E. Miller

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

7/31/89
Date

Inspection Summary
Inspection during the period of May 30 through July 14, 1989 Report No. 50-010/89002(DRP); No. 50-237/89017(DRP); No. 50-249/89016(DRP))
Areas Inspected: Routine unannounced resident inspection of previously identified inspection items, license events reports followup, allegations followup, plant operations, maintenance and surveillances, safety assessment/ quality verification, radiological controls, engineering/technical support, Dresden Station management organization and report review.
Results: One violation was identified during this inspection period concerning the Unit 2 excessive drywell temperature event of October 29, 1988 (Paragraph 11).

- During this inspection period, one reactor scram occurred from power. This one scram was attributed to drifting main steamline temperature switches during a surveillance test.

DETAILS

1. Persons Contacted

Commonwealth Edison Company (CECo)

*E. Eenigenburg, Station Manager
*J. Kotowski, Production Superintendent
*L. Gerner, Technical Superintendent
C. Allen, Administrative Service Superintendent
D. Van Pelt, Assistant Superintendent - Maintenance
G. Smith, Assistant Superintendent - Operations
B. Zank, Operating Engineer
K. Peterman, Regulatory Assurance Supervisor
W. Pietryga, Operating Engineer
J. Achterberg, Technical Staff Supervisor
L. Johnson, Q.C. Supervisor
J. Mayer, Station Security Administrator
D. Morey, Chemistry Services Supervisor
D. Saccomando, Health Physics Services Supervisor
*K. Kociuba, Q.A. Superintendent
*R. Falbo, Regulatory Assurance Group Leader

The inspectors also talked with and interviewed several other licensee employees, including members of the technical and engineering staffs, reactor and auxiliary operators, shift engineers and foremen, electrical, mechanical and instrument personnel, and contract security personnel.

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

2. Previously Identified Inspection Items (92701 and 92702)

(Closed) Open Item (237/88012-01 and 249/88014-01): Review calculations to validate drywell spray initiation pressure limit curve. During the review of Dresden Emergency Operating Procedures, (EOP) the inspectors requested and could not be provided the calculations to validate the 5 psid differential pressure limit between torus and drywell for initiation of drywell sprays. The licensee found an evaluation for Pilgrim Station Mark I Containment, that is similar to Dresden's, which verified a safe torus to drywell differential pressure capability of 8 psid. The licensee later calculated a site specific limit of 8.3 psid as the maximum allowable negative pressure differential between the drywell and torus with a positive torus pressure. These items are considered to be closed.

(Closed) Open Item (237/88012-02 and 249/88014-02): Review justification for using 200 degrees as an entry condition for primary containment high temperature. The use of a 200 degree F temperature limit for entry into the Primary Containment Control EOP, which is greater than the maximum normal operating average temperature

specified by the Boiling Water Reactor Owners Group (BWROG) Emergency Procedure Guidelines (EPG), is justified because of the location of some of the thermocouples close to equipment with high temperatures during operation. The entry condition for Primary Containment Control remained after the licensee's engineering evaluation, at 200 degrees F as indicated on any one of the five thermocouples specified.

(Closed) Open Item (237/88012-03 and 249/88014-03): Review calculations showing transition from torus to drywell pressure used to create nomograph showing allowable pump Net Positive Suction Head (NPSH). A review of the data used to develop the Emergency Core Cooling System (ECCS) suction nomographs, which are provided in Dresden Emergency Operating Procedure (DEOP) 010 to protect the ECCS pumps from cavitation, indicated that the correlation between Drywell and Torus pressure could be incorrect when torus water level was above 11 feet. The ECCS suction nomograph was revised on October 27, 1988, to use the newly installed Torus Bottom Pressure Indication which indicates from 0 to 100 psig. These items are considered to be closed because of the use of the bottom pressure indication.

(Closed) Violation (237/87040-01): Previous corrective actions failed to prevent a repetitive violation. This violation involved the by-passing of more Average Power Range Monitor (APRM) channels than permitted by Technical Specifications and was similar to a previous violation (50-237/87026-01) where the number of Reactor Protection System (RPS) Channel B APRM/ Intermediate Range Monitor (IRM) companion trip functions had been reduced to only one. The root cause of this violation was attributed to a personnel error. The inspector reviewed the following corrective actions taken to prevent recurrence:

- Precautions were added to procedures along with a table illustrating the IRM/APRM companion relationship.
- A placard was added to the panel board listing the IRM/APRM complements.
- A procedure change to Appendix A, Shift Turnover, requires a check of the IRM/APRM configuration each shift turn over.
- Operator training has been completed.
- A Technical Specification Amendment was requested and issued, Amendments 237/100 and 249/96 on August 24, 1988, to eliminate the APRM downscale trip requirements.
- A cover has been placed over the IRM/APRM bypass joysticks as a reminder to assure proper configuration prior to bypassing an APRM or IRM.

- A letter to all licensed personnel reviewed the event and emphasized the importance of joystick configuration.

(Closed) Unresolved Item (249/86012-30): Safety System Outage Modification Inspection (SSOMI) Unresolved Item 2.4-2, Seismic Qualification of LPCI Room Cooler Motors. This unresolved item concerned the adequacy of seismic qualifications for Westinghouse motors used for operation of the LPCI room coolers. In question was the application of the rigid mount criteria used in the original seismic qualification to the flexible motor mount that is used in the field installation. Also see Inspection Report 50-249/88200.

The licensee had obtained calculations for the LPCI room cooler fan motor mounting configuration which confirmed that the LPCI fan motors are seismically qualified as installed. The unresolved item is considered to be closed because of the licensee obtaining a recent seismic qualification for the actually installed fan motor.

3. Licensee Event Reports (LER) Followup (92700)

Through direct observations, discussions with licensee personnel, and review of records, the following event reports were reviewed to determine that reportability requirements were fulfilled, immediate corrective action was accomplished, and corrective action to prevent recurrence had been accomplished or scheduled in accordance with Technical Specifications.

(Closed) LER 249/89003: Spurious Group V Primary Containment Isolation While Shutdown Due to a Design Deficiency. With Unit 3 shutdown for a scheduled maintenance outage, an unexpected Group V isolation occurred resulting in the isolation of the isolation condenser. This event was attributed to differential pressure spikes and/or noise generated by an annubar flow instrument. This instrument was installed on the isolation condenser condensate return line to replace the previous elbow type instrument in 1985. Due to three previous occurrences, the last of which occurred on August 7, 1987, time delays were installed in the isolation circuitry. Because of a problem with setpoint drift on these time delay relays, a modification has been initiated to install relays with a shorter time delay during the next refuel outage.

(Closed) LER 249/89007: Primary Containment Personnel Access Hatch Local Leak Rate Test (LLRT) Failure. With Unit 3 at 21% rated thermal power following a scheduled maintenance outage, a drywell personnel access airlock failed its local leak rate test. The reactor was shutdown and primary containment de-inerted to facilitate further inspection. The licensee investigation revealed that the airlock inner door gasket seal had not seated evenly in the gasket grove causing the seal to be pinched through repeated usage of the airlock during the outage. This resulted in a six inch longitudinal tear in the gasket seal. The local leak rate test was successfully completed following replacement of the gasket seal. A revision is to

be submitted for Dresden Operating Surveillance (DOS) 1600-10, Pre-Startup Drywell Inspection Plan, to include a detailed inspection checklist to ensure proper seating of the gasket seal prior to final closing. The preventative maintenance program is also to be revised to require the gasket to be replaced every refueling outage.

*(Closed) LER 237/89016: High Pressure Coolant Injection (HPCI) Piping Found in Violation of Final Safety Analysis Report (FSAR) Design Criteria due to Management Deficiency. Through a licensee HPCI Safety System Functional Inspection (SSFI) and subsequent analysis, it was determined that the Unit 2 and 3 HPCI turbine steam supply valves 2(3)-2301-3 drain pot piping did not meet FSAR design criteria. Further analysis showed that the piping would, however, remain operable under all design basis events. The licensee attributed this event to modifications performed in 1982 without the benefit of a formal thermal and seismic analysis. Although the Boiling Water Reactor Engineering Department (BWRED) had previously become aware of the design discrepancies as early as September 1984, Station personnel were not notified since it was believed the problem would be corrected through a pending modification. However, the pending modification was subsequently cancelled, and BWRED was not notified of the cancellation. Because of this, the design discrepancies remained until they were recently re-identified by the licensee's HPCI SSFI. Additional supports were subsequently added to Unit 3 and similar work is currently ongoing for Unit 2. The modification program was previously upgraded through a revision to Dresden Administrative Procedure (DAP) 5-1 Plant Modification Program. This included modification cancellation instructions requiring notification of affected station departments, Nuclear Licensing, BWRED and the designer. More detailed administrative controls on modifications were also delineated, as well as a design walkdown checklist used to confirm conceptual design and to provide input into the detailed design. In addition, BWRED is currently developing a procedure to give guidance when an analysis finds equipment in conflict with the FSAR. The previous inadequate design controls have been identified as an licensee identified violation (237/89017-01) and is considered closed in this report due to adequately completed or planned corrective actions meeting the criteria of 10 CFR 2 listed below.

*(Closed) LER 249/89008: Fire Damper Discovered Obstructed by Welding Equipment Due to Management deficiency. The description of this event, including licensee investigation and corrective actions are described in Paragraph 6.c.3 of this report.

*Note: The preceding LERs have been reviewed against the criteria of 10 CFR 2, Appendix C, and the incidents described meet all of the following requirements. Thus no Notice of Violation is being issued for these items.

a. The event was identified by the licensee,

b. The event was an incident that, according to the current enforcement policy, met the criteria for Severity Levels IV or V violations,

c. The event was appropriately reported,

d. The event was or will be corrected (including measures to prevent recurrence within a reasonable amount of time), and

e. the event was not a violation that could have been prevented by the licensee's corrective actions for a previous violation.

No violations or deviations, other than the noted licensee identified, were identified in this area.

4. Allegations Followup (AMS No. RIII-89-A-0044) (Closed)

On March 20, 1989, the Region III duty officer received a telephone call from an individual who expressed concerns related to leaks in the Unit 2 offgas system during late February and early March 1989. The caller would not provide his name.

During this inspection, the inspector reviewed licensee records and reports and interviewed licensee and contractor personnel to determine the validity and consequences of the concerns expressed by the alleger. The allegations are described and discussed below.

Allegation: Plant management was not very concerned about a leak in the offgas rooms which resulted in several personal contaminations.

Discussion: According to licensee personnel and records, on February 25, 1989, shortly after startup of Unit 2 after a refueling and maintenance outage, the clothing of some personnel on the 549-foot level of the turbine building was becoming contaminated with short-lived particulate daughters of noble gases. On February 25 and 26, the licensee found some problems with offgas recombiner fans and fan doors; these possible sources of the offgas leaks were repaired.

After the problem began on February 25, the licensee collected particulate air samples near the steam jet air ejector rooms on a shift and/or daily basis dependent on air activity levels. The particulate air activity was found to increase periodically but at no set frequency. The air activity was always a small fraction of isotopic maximum permissible concentrations and displayed a half-life of about thirty minutes.

On March 1, 1989, licensee radiation protection personnel performed radiation surveys on the hydrogen addition systems and found a valve which was leaking. The Unit 2 shift foreman was informed of the leak. The leaking valve, however, was not repaired until March 8, 1989. After repair of the leak, air samples no longer identified elevated short-lived particulate activity.

On March 16, 1989, the licensee again began to experience increased airborne particulate activity in the same general area. On March 20, 1989, the licensee again performed radiation surveys on the hydrogen addition system; no leaks were found. During review of airborne particulate air sample results, the Unit 2 Radiation Protection (RP) Foreman noted that the airborne particulate activity was elevated when the hydrogen addition system was on, and low or nonexistent when the hydrogen system was off. The RP foreman reported this information to the shift foreman for operations who had an operator check the valving lineup between the hydrogen addition and hydrogen monitoring system. The operator found and corrected a valve alignment problem. No further problem with airborne particulate activity was experienced.

During review of this matter, the inspector learned that work was intermittently in progress to perform a modification of the hydrogen monitoring system. It appears that there was more than one source of offgas leakage during the period February 25 through March 20, 1989, and work on the hydrogen monitor contributed to the offgas leaking problem.

Finding: The allegation/concern was partially correct in that an offgas leak was identified on March 1, 1989, which was not repaired until March 8, 1989. However, no licensee procedure or policy, or regulatory requirement, was violated. The leak did not pose a significant radiological hazard to station personnel.

Allegation: Lung dose to workers from airborne radioactivity is unknown.

Discussion: 10 CFR 20.103(a)(1)Note 2 allows individual exposures to noble gases and their daughters to be accounted for as part of the limitation on individual external doses because the Maximum Permissible Concentrations (MPCs) listed in Table 1 Column 1 are based on exposure to the material as an external radiation source. Therefore, it is not necessary to make an additional determination of lung dose for exposure to these nuclides.

Finding: The allegation/concern was not substantiated.

5. Plant Operations (71710, 71707 and 93702)

   a. Enforcement History

   During this inspection period, no violations or deviations were identified in the plant operations functional area. However, one item which occurred in a previous inspection period dealing with the high drywell temperature event of October 28, 1988, was determined to be a violation as described in Paragraph 11.b of this report.

b. Operational Events

(1) On July 10, 1989, with Unit 2 at 63% rated thermal power, Recirculation Pump A speed unexpectedly drifted downward causing about a 3% decrease in both total core flow and reactor power. This caused the plant to enter the instability region of the power to flow map. The operators manually locked up the Recirculation Pump scoop tube to stop the speed drift and inserted CRAM arrays to exit the instability region. Recirculation Pump speeds were then matched by manual hand cranking of the scoop tube for Recirculation Pump A. Specific maintenance activities associated with this event are described in Paragraph 6.c.5 of this report.

(2) On July 12, 1989, Unit 2 received a reactor scram on a spurious Main Steam Line (MSL) High temperature signal RPS Channel B while Channel A was in a half scram condition during surveillance testing. The operators were able to achieve pressure control with the reactor water cleanup system (due to low decay heat) and vessel level was maintained within the normal operating range. Specific maintenance activities associated with this event are described in Paragraph 6.c.4 of this report.

c. Observation of Operations

The inspectors observed control room operations, reviewed applicable logs and conducted discussions with control room operators during this period. The inspectors verified the operability of selected emergency systems, reviewed tagout records and verified proper return to service of affected components. Tours of the Unit 2 and 3 reactor buildings were conducted to observe plant equipment conditions, including potential fire hazards, fluid leaks, and excessive vibrations and to verify that maintenance requests had been initiated for equipment in need of maintenance.

The inspectors, by observation and direct interview, verified that the physical security plan was being implemented in accordance with the station security plan.

The inspectors reviewed new procedures and changes to procedures that were implemented during the inspection period. The review consisted of a verification for accuracy, correctness, and compliance with regulatory requirements.

The inspectors also witnessed portions of the radioactive waste system controls associated with radwaste shipments and barreling.

These reviews and observations were conducted to verify that facility operations were in conformance with the requirements established under technical specifications, 10 CFR, and administrative procedures.

d. Engineered Safety Features (ESF) System Walkdown

The inspector walked down the accessible portions of the Units 2 and 3 Standby Gas Treatment System (SGTS) to verify operability by comparing system lineup with plant drawings, as-built configuration, and operations checklists; observing equipment that could degrade performance; and verifying that instrumentation was properly valved, functioning, and calibrated. The inspectors also observed plant housekeeping/cleanliness conditions and radiation protection practices.

The inspector noted several discrepancies between the as-built configuration and plant drawing M-49. This included differences in the locations of specific temperature indicators, incorrect numbering of temperature indicators on the drawing and a pressure indicator installed in place of a temperature indicator shown on the drawing. All of these discrepancies were discussed with the System Engineer for resolution.

No violations or deviations were identified in this area except as described in Paragraph 11.b of this report.

6. Maintenance and Surveillances (62703, 61726, and 93702)

The inspectors performed the following:

a. Unit 1

As general background, Unit 1 was shutdown in the late seventies and was never restarted after Three Mile Island (TMI) because of the costs associated with bringing the facility into conformance with post TMI safety requirements. All fuel elements and control rods were removed and stored in the fuel storage pool. The primary system was thoroughly chemically cleaned.

The licensee proposed that Unit 1 would remain in this SAFSTOR condition until Units 2 and 3 are shutdown for decommissioning and submitted a SAFSTOR Decommissioning Plan and associated Technical Specifications (TS) for this period. These proposals are presently under review by NRC.

In the course of the review of the proposed TS surveillance program for Unit 1, an inspection of the present surveillance program required by the existing TS was conducted. Since Unit 1 is in the shutdown defueled condition described above, operational surveillance requirements are no longer necessary such as safety limits, limiting

safety system settings and most of the limiting conditions for operation (LCO). However, the licensee still conducts LCO TS surveillances on radiological materials (airborne and liquid effluents, waste storage and environmental monitoring), storage fuel pool water level, fire protection systems and auxiliary electrical system batteries.

The inspector reviewed surveillance procedures, check-sheets and schedules to verify that all TS required surveillances for Unit 1 were being conducted at the required frequencies.

In addition, the inspection also reviewed the other safety and preventive maintenance (PM) checks contained in the Unit 1 General Surveillance System Master File and required by TS to ascertain that these checks were scheduled and completed. These surveillances and PM items numbered 190 and included such areas as instrument, Area Radiation Monitor (ARM) and gauge calibrations, routine radiation and contamination surveys, boiler and pressurized vessel inspections, fuel pool structure and fuel assembly conditions, fuel inventory, crane and hoist inspections, inspection and lubrication of pumps, valves, blowers, compressors, traveling screens, and emergency lighting. The inspector concluded that the instrumentation and equipment necessary to safely maintain Unit 1 in the SAFSTOR mode were listed in the Master File and the surveillances and services were properly being conducted as scheduled.

b. Units 2 and 3

The inspectors observed surveillance testing required by technical specifications for the items listed below and verified that testing was performed in accordance with adequate procedures, that test instrumentation was calibrated and that limiting conditions for operation were met. The inspectors also verified that removal and restoration of the affected components were accomplished, that test results conformed with technical specifications and procedure requirements and were reviewed by personnel other than the individual directing the test, and that any deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel.

The inspectors witnessed portions of the following test activities:

Unit 2

Average Power Range Monitor/Rod Block Monitor Flow Converter to Total Core Flow Adjustment
Standby Liquid Control System Pump Test
Quarterly Standby Liquid Control Pump Test for Inservice Testing Program

Unit 3

Core Spray System Pump Test
Core Spray System Valve Operability check
Low Pressure Coolant Injection System Pump Operability Test
Suppression Chamber to Drywell Vacuum Breaker Full Stroke Exercises
Rod Block Monitor Functional Testing

Units 2 and 3

HPCI System Operability Verification
Reactor Low Water Level Scram and Low Low Water Level Isolation Trip
HPCI Steam Line High Flow
Isolation Trip
HPCI Turbine Permissive

Station maintenance activities of safety related systems and components listed below were observed/reviewed to ascertain that they were conducted in accordance with approved procedures, regulatory guides and industry codes or standards and in conformance with technical specifications.

The following items were considered during this review:

The limiting conditions for operation were met while components or systems were removed from service; approvals were obtained prior to initiating the work; and activities were accomplished using approved procedures and were inspected as applicable. Functional testing and/or calibrations were performed prior to returning components or systems to service; quality control records were maintained; activities were accomplished by qualified personnel; and parts and materials used were properly certified. Radiological and fire protection controls were properly implemented. Work requests were reviewed to determine status of outstanding jobs and to assure that priority is assigned to safety related equipment maintenance which may affect system performance.

Various maintenance activities associated with the following events were observed/reviewed.

c. Operational Events

(1) On May 31, 1989, during the Unit 3 startup following the main transformer replacement outage, a controlled shutdown was conducted to perform repairs on a primary containment inner interlock door. This event is discussed in more detail in Paragraph 3 of this report.

(2) On June 13, 1989, the Unit 2 Reactor Feedwater Pump (RFP) C Outboard Seal was discovered to have failed. RFP B was already out of service due to a previous leak in it's discharge check valve. RFP A although operating, had a

small leak in it's suction valve. Removal of RFP C from service necessitated a power reduction from about 500 MWe to 280 MWe to stay well within the capabilities of the remaining RFP A.

(3) On June 17, 1989, while performing rounds, a Shift Foreman found two vertical fire dampers in a Unit 3 HPCI room non-ducted ventilation opening blocked open with an air hose and a welding lead. These obstacles were immediately removed such that the fire dampers were returned to operability. The licensee determined that these obstructions were routed through the fire barrier on June 14, 1989, while maintenance was being performed in the HPCI room. Dresden Technical Specification 3.12.F.2 requires that a continuous fire watch be established within one hour when a penetration fire barrier protecting safety related areas is not intact and equipment on either side of the barrier is required to be operable. Since Unit 3 was at power during that time and thus the HPCI pump on one side and various low pressure ECCS pumps on the other side of the fire barrier were required to be operable, the failure to establish a continuous fire watch constitutes a violation of Technical Specifications.

This event has been reviewed against the criteria of 10 CFR 2, Appendix C, and the incident described meets all the requirements described in the note in Paragraph 3 of this report. Thus, no Notice of Violation is being issued for this item (249/89016-01) and this item is considered closed.

(4) On June 23, 1989, with Unit 3 at 98% rated thermal power, a small flash fire occurred in the Main Generator Core Monitor. The licensee believed that a small hydrogen leak in the Core Monitor led to an excessive hydrogen concentration and subsequent detonation while an Instrument Technician was preforming maintenance. The fire was only momentary and a fire extinguisher was immediately used to further ensure that the fire was out. No damage was visually apparent and no injuries occurred.

(5) On July 7, 1989, with Unit 2 at 72% rated thermal power, the Reactor Building Ventilation System isolated and the Standby Gas Treatment System actuated during a Reactor Building Ventilation Radiation Monitor functional surveillance test. When the controller for Radiation Monitor B was pulled from its panel to conduct the surveillance a nicked wire shorted against the chassis causing a spike on the channel. The exposed wire was temporarily taped pending scheduling of permanent repairs.

(6) On July 12, 1989, Unit 2 received a reactor scram on Main Steam Isolation Valve (MSIV) closure during a surveillance on the RPS B power supply. The RPS B motor generator supplies the A channels of RPS and Group 1 Primary Containment Isolation System (PCIS). During the surveillance, a half scram and half isolation was received on RPS Channel A and PCIS Channel A per the procedure. However, a spurious Main Steam Line (MSL) High Temperature Signal was received on Channel B prior to resetting the surveillance induced half scram and half isolation signals. This resulted in a full Group 1 isolation (MSIVs closing) and a resulting reactor scram. All systems responded as expected and no safety systems actuated.

(7) The inspectors observed completion of a work request involving Unit 2 APRM flow-biased scram, rod block and downscale calibrations. This work conducted was the result of failure of an APRM rod block functional surveillance test.

d. Approach to the Identification and Resolution of Technical Issues From a Safety Standpoint

(1) A power reduction on Unit 2 to 300 MWe was held on June 9, 1989. This was to facilitate a drywell entry in order to complete repairs on a Traversing Incore Probe (TIP) machine. The unit was also placed in single loop operation to facilitate repair of Recirculation Pump A Motor Generator Set outboard bearings. Other Unit 2 activities that occurred during the power reduction included investigation of spurious oscillations received on Turbine Control Valve #1, Main Steam Isolation Valve (MSIV) timing and replacement of cards in the Feedwater Level Control System panels.

(2) Following the removal of Unit 2 RFP C from service on June 13, 1989, RFP B was restarted later in the day with the discharge check valve leaking 2-3 gpm and reactor power was increased as requested by the load dispatcher. The leak in the RFP A suction valve and the failed RFP C outboard seal were subsequently repaired. RFP B discharge check valve still leaked as of the end of the inspection period.

(3) Following the Unit 2 scram of July 12, 1989, the temperature switches (which had experienced drift) associated with MSL Channel B detectors were replaced and the plant commenced startup on July 13, 1989.

(5) Following the drifting of Recirculation Pump 2A speed on July 10, 1989 (see Paragraph 5.b of this report), the licensee replaced the Recirculation Pump Motor-Generator (M-G) set tachometer which was sending incorrect signals to the velocity feedback circuitry. In addition, a milli volt/amp converter was replaced in the velocity feedback circuitry.

e. Responsiveness to NRC Initiatives

The inspector expressed a concern to the licensee about an upward trend in control room work requests as early as February 1989. Although initial licensee action was delayed as to this concern, the licensee began to address this issue in May 1989. The licensee's investigation found that the work analysts were not recognizing these as affecting the control room and, as such, the priorities assigned to these work activities were too low.

7. Radiological Controls (92702)

Operational Events

- On June 8, 1989, the May 1989 Batch Waste Release Tank composite sample for tritium and gross alpha was inadvertently discarded before it could be sent offsite for analysis. Further review by the NRC is required, this is considered an Unresolved Item (237/89017-02).

- On June 15, 1989, while performing a Quality Assurance walkdown of owner controlled property outside the protected area, the licensee found 145 55-gallon drums, some bearing visible low specific activity markings, in an old dumpsite. It appeared that the markings had at one time been painted or taped over to obscure them. The licensee surveyed the drums as they were removed from the dumpsite. Three of these drums and a concrete liner also discovered in the dumpsite were found to have low levels of radioactive contamination. These levels included 1.2k, 30k and 60k disintegrations per minute (DPM) on each of the three drums, respectively, and 300k DPM on the concrete liner. The licensee indicated that the materials were placed in the dumpsite prior to 1981 and that these contamination levels were too low to detect with instrumentation available at that time.

  The licensee removed the empty noncontaminated drums for general disposal. Drums with contamination were removed and stored in the radwaste area. Approximately 23 drums containing liquid or solid residue were also stored pending chemical analysis.

  An NRC inspector was dispatched to the site on June 22, 1989 to verify the licensee's findings and observe some of the licensee's radiological surveys. The inspectors agreed with the licensee's findings and corrective actions.

No violations or deviations were identified in this area.

8. Safety Assessment/Quality Verification (40500)

a. The inspector observed a licensee training session pertaining to the history of the counterfeit molded-case circuit breaker (MCCB) issue and Nuclear Management and Resources Council (NUMARC) visual inspection guidelines. This training, as described in the letter, M. H. Richter to U. S. NRC, dated July 7, 1989, was a result of licensee participation in a NUMARC industry initiative to ensure reliable performance of MCCBs used in non-safety related applications. The training was conducted prior to performing a visual inspection of the non-safety related MCCB inventory.

b. The inspector attended the licensee's June 1989 monthly performance review meeting. In addition to discussions involving the plant status and activities for the previous month, each of the plant's top ten technical issues as determined by the licensee were reviewed. A summary of performance during the Unit 3 transformer outage and activities of the Scram/Engineered Safety Feature Actuation Reduction Committee were also discussed. Particular management concern relating to an increase in Control Room work requests was expressed. The licensee conducted an evaluation to determine the cause of this increase. The results of the evaluation are discussed in Paragraph 6.d.2 of this report.

No violations or deviations were identified in this area.

9. Engineering/Technical Support (37700)

a. Approach to the Identification and Resolution of Technical Issues From a Safety Standpoint

The inspector reviewed Partial Modification Design Package, M12-2/3-87-05C, Control Room Modifications, one of 10 packages concerning the consolidation of the Unit 1 Control room into the Unit 2/3 Control room and the utilization of the control room as additional office space. All unnecessary Unit 1 Control room panels and instrumentation will be removed and a new seismically designed three-hour rated fire wall and security barrier will be installed to separate the Unit 2/3 Control room from the new office space. The subject partial modification concerns the installation and testing of a new Process/Meteorology/Radiation Panel, 901-2. Except for existing Panels 18 and 18C, Electrical Switchyard Control and Instrumentation, and Panel B-1, Station Auxiliary Power Control and Indication, which will be retained intact; all necessary Unit 1 instrumentation and controls will be consolidated into the new panel. All of the above panels, along with new kitchen-eating facilities and locker room-toliet facilities, will be located within the Unit 2/3 Control room.

Instrumentation and controls for the new 901-2 Panel will include:

1. A new ARM recorder.
2. A new service water discharge monitor.
3. A new annunciator panel for Unit 1 systems.
4. Connecting existing ARMs to the new recorder-annunciator.
5. Relocation of controls, indications and trouble annunciations from the old Unit 1 panels for service water system, bearing lube water system, turbine building closed cooling water system, fire pump discharge pressure, screen wash pumps, condenser circulating water pumps, well water system, clean demineralized water tank, contaminated water makeup, instrument air system, service air system, meterological data (2 recorders), and other trouble annunciators such as sphere and turbine building ventilation, radwaste building, instrument air dryer, sphere drain tank high level, heating system boiler and fuel pool high and low level.

Many of the above system indication relocations will include new pressure transmitters and transmitter power supplies. The modification will also require relocation of existing facilities such as breathing air piping, control room penetrations, HVAC system ducts, and electrical-telephone systems.

The inspector reviewed the partial modification package to verify that all systems depicted in the Unit 1 Decommissioning Plan were included, that all new and relocated instruments, annunciators, and controls would be calibrated and tested following the modification and before use, and that a 10 CFR 50.59 review had been completed and approved.

b. Responsiveness to NRC Initiatives

The licensee was particularly responsive to providing answers to questions on various technical issues requested by the NRC regional office. These areas included plant specific testing of diesel generator trips and bypasses and the source of RPS response times used in reload safety evaluations.

No violations or deviations were identified in this area.

10. Dresden Station Management Organization

During this inspection period, CECo announced several key management changes including the following:

C. Schroeder, Technical Superintendent to Corporate Outage Planning

L. Gerner, Production Superintendent to Technical Superintendent

J. Kotowski, Assistant Superintendent-Operations to Production Superintendent

G. Smith, Operating Engineer to Assistant Superintendent-Operations

11. Report Review

a. During the inspection period, the inspectors reviewed the licensee's Monthly Operating Report for June. The inspectors confirmed that the information provided met the requirements of Technical Specification 6.6.A.3 and Regulatory Guide 1.16.

b. The inspectors completed the review of the Dresden Unit 2 Drywell Temperature Event Evaluation Report prepared by Commonwealth Edison Company and Sargent and Lundy for the October 29, 1988 event. This event was previously discussed in Inspection Reports 50/237/88026; 50/249/88026 and 50-237/89011; 50-249/89010. The licensee attributed this event to the absence of cooling airflow to the reactor head area due to the ventilation hatches, provided in the bulkhead plate, being left in the closed position. The licensee determined the primary root cause to be inadequacies in procedures which direct operations and maintenance personnel to open the hatches and perform an inspection prior to startup. Although Dresden Maintenance Procedure (DMP) 1600-5, Drywell Head Replacement and Installation of Shield Blocks, Revision 2, contained a step to open all required ventilation openings, it did not clearly identify which hatches were required to be open. Only the manway hatches were found open. In addition, Dresden Operating Surveillance Procedure (DOS) 1600-10, Pre-Startup Drywell Inspection Plan, Revision 4, which contains a step to verify that the hatch doors to the reactor head area are open, was misinterpreted by the shift supervisors who made the inspection as applying only to the manway hatches. Consequently, ventilation hatches were not checked. The inadequate procedures to which this event was attributed are considered to be a violation of Technical Specification 6.2.A (237/89017-03).

The licensee identified and implemented extensive corrective actions in response to this event and, as such, this item is considered closed.

These corrective actions included the following:

- Revisions of the inadequate procedures.
- Evaluation of the remaining life of environmentally qualified equipment.
- Repair and replacement of electrical and mechanical equipment and cables as required.
- Installation of an upgraded drywell temperature monitoring system.
- Repair of the drywell cooling system and conduct of a performance test.

- Repair of thermal insulation as required.
- Repaint of the drywell dome and scraping of other drywell surfaces to remove loose paint.
- Implementation of a procedure for monitoring and elevating drywell thermocouple data.
- Performance of a drywell insulation system evaluation.
- Review and update of equipment qualification binders as necessary.

No other violations or deviations were identified.

12. Exit Interview (30703)

The inspectors met with licensee representatives (denoted in Paragraph 1) on July 13, 1989, formally 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 inspector during the inspection. The licensee did not identify any such documents/processes as proprietary. The licensee acknowledged the findings of the inspection.

Tab 13

DRESDEN 2 & 3

FIRE PROTECTION PROGRAM DOCUMENTATION PACKAGE

Inspection Report No. 50-237/89022 and 50-249/89021

89022/89021

Original: File
cc: E. Eenigenburg
[illegible]
[illegible] Stoltz
M. Strait,
R. Tello
R. Stobert/M. Dillon
L Assist
R. Johnson/B. Barth
L Assist
R. Geier
L Assist
L Prepare draft Response 1/12/[illegible]

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

Revision 8
April 1992

[illegible] 28

DEC 26 1989

Docket No. 50-237
Docket No. 50-249

Commonwealth Edison Company
ATTN: Mr. Cordell Reed
Senior Vice President
Post Office Box 767
Chicago, IL 60690

Note to VanPelt: in the NOV write up, it makes reference to repeat failures of MM personnel to recognize fire barriers. This should be addressed in the response.

Gentlemen:

This refers to the routine safety inspection conducted by S. G. Du Pont and D. E. Hills of this office on October 11 through December 1, 1989, of activities at Dresden Nuclear Power Station, Units 2 and 3, authorized by NRC Operating Licenses No. DPR-19 and No. DPR-25 and to the discussion of our findings with Mr. E. Eenigenburg and others at the conclusion of the inspection.

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.

During this inspection, certain of your activities appeared to be in violation of NRC requirements, as described in the enclosed Notice. A written response is required. In accordance with 10 CFR 2.790, of the Commission's regulations, a copy of this letter and the enclosures will be placed in the NRC Public Document Room.

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

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

Sincerely,

W D Shafer

W. D. Shafer, Chief
Reactor Projects Branch 1

Enclosures:
1. Notice of Violation
2. Inspection Report
   No. 50-237/89022(DRP)
   No. 50-249/89021(DRP)

See Attached Distribution

NOTICE OF VIOLATION

Commonwealth Edison Company
Dresden Nuclear Station

Docket No. 50-237
Docket No. 50-249

As a result of the inspection conducted on October 11 thru December 1, 1989, and in accordance with the General Policy and Procedures for NRC Enforcement Actions, (10 CFR Part 2, Appendix C), (1989) the following violation was identified:

1. 10 CFR 50.48(a) requires that each operating nuclear power plant have a fire protection plan that satisfies Criterion 3 of Appendix A to 10 CFR Part 50. It further requires that the plan shall describe specific features necessary to implement the plan such as administrative controls to limit fire damage to structures, systems or components important to safety so that the capability to safely shut down the plant is assured.

   Section C.1 of the licensee's response to the Guidelines of Appendix A to Branch Technical Position APCSB 9.5-1 as accepted in the 1980 Supplemental Safety Evaluation Report indicates that administrative measures are established to ensure that guidelines of the Branch Technical Position are included in design and procurement documents and that deviations therefrom are controlled.

   Contrary to the above, a penetration in a three hour fire rated wall located in a safety related area of the 570 feet elevation of the reactor building, as prescribed by Section D.1.j of the Branch Technical Position, was not included in design documents and deviations were not controlled. The fire rated wall was degraded in 1985 by replacement of the original piping with non-approved polyvinyl chloride plastic piping and was further degraded on October 25, 1989 when the piping was completely removed and the penetration left unsealed.

   This is a Severity Level IV violation (Supplement I) (No. 237/89022-02(DRP)).

Pursuant to the provisions of 10 CFR 2.201, you are required to submit to this office within thirty days of the date of this Notice a written statement or explanation in reply, including for each violation: (1) corrective action taken and the results achieved; (2) corrective action to be taken to avoid further violations; and (3) the date when full compliance will be achieved. Consideration may be given to extending your response time for good cause shown.

DEC 2 6 1989
Date

W D Shafer
W. D. Shafer, Chief
Reactor Projects Branch 1

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-237/89022(DRP); No. 50-249/89021(DRP)

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

Licensee: Commonwealth Edison Company
P. O. Box 767
Chicago, IL 60690

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

Inspection At: Dresden Site, Morris, Illinois

Inspection Conducted: October 11 through December 1, 1989

Inspectors: S. G. Du Pont
D. E. Hills

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

12/26/89
Date

Inspection Summary

Inspection during the period of October 11 through December 1, 1989 (Report No. 50-237/89022(DRP); No. 50-249/89021(DRP))
Areas Inspected: Routine unannounced safety inspection by resident inspectors of previously identified inspection items, licensee event reports, plant operations, maintenance and surveillance, safety assessment/qualify verification, engineering/technical support and report review.
Results:

- Specific events demonstrating management involvement and a regard for correctly meeting requirements as well as for minimizing unplanned transients were noted.

- One violation was identified during the inspection period as described in Paragraph 5.b.8. This involved the failure to properly control the design of a penetration through a fire barrier such that maintenance personnel degraded that barrier on two separate occasions. This specific event was considered to be of minimum safety significance although a previous degradation of a fire barrier by maintenance personnel was documented in a previous inspection report. This was not considered to be indicative of what are usually thorough and effective corrective actions by the licensee.

Distribution

cc w/enclosures:
T. Kovach, Nuclear
Licensing Manager
E. D. Eenigenburg, Station Manager
DCD/DCB (RIDS)
Licensing Fee Management Branch
Richard Hubbard
J. W. McCaffrey, Chief, Public
Utilities Division
LaSalle RIO
Quad Cities RIO
Dresden RIO

- Two unresolved items were identified in Paragraphs 5.b.5 and 7.b.3. One involved whether adequate corrective actions were taken in response to previously identified HPCI piping support discrepancies. The other involved installation of main steamline leak detection temperature switches without the appropriate environmental qualification documentation.

DETAILS

1. Persons Contacted

Commonwealth Edison Company

*E. Eenigenburg, Station Manager
*L. Gerner, Technical Superintendent
E. Mantel, Services Director
*J. Kotowski, Production Superintendent
D. Van Pelt, Assistant Superintendent, Maintenance
J. Achterberg, Assistant Superintendent, Work Planning
*G. Smith, Assistant Superintendent, Operations
*K. Peterman, Regulatory Assurance Supervisor
*C. Allen, Performance Improvement Supervisor
W. Pietryga, Operating Engineer
*R. Stobert, Operating Engineer
M. Korchynsky, Operating Engineer
B. Zank, Operating Engineer
J. Williams, Operating Engineer
*M. Strait, Technical Staff Supervisor
L. Johnson, Q.C. Supervisor
J. Mayer, Station Security Administrator
*D. Morey, Chemistry Services Supervisor
*D. Saccomando, Health Physics Services Supervisor
E. Netzel, Q.A. Superintendent
*R. Falbo, Regulatory Assurance Group Leader
K. Yates, Nuclear Safety Supervisor
*K. Kociuba, Quality Assurance Superintendent

The inspectors also talked with and interviewed several other licensee employees, including members of the technical and engineering staffs, reactor and auxiliary operators, shift engineers and foremen, electrical, mechanical and instrument personnel, and contract security personnel.

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

2. Previously Identified Inspection Items (92701 and 92702)

(Open) Open Item (No. 249/89011-02): The licensee was to provide a written response describing planned corrective actions to ensure that usage of the isolation condenser for extended time periods without offsite power would not result in radioactive releases. The latest response to this issue by the licensee was contained in the letter from J. A. Silady to A. B. Davis dated November 15, 1989. A tentative schedule was established for the respective unit refueling outages at the end of Cycle 13 in 1992 to install diesel driven pumps for supply of clean demineralized water to the shell side of the isolation condensers from the clean demineralized water storage tank. A proposed design improvement to supply 480 VAC power to the isolation condenser shell side motor-operated clean demineralized water fill valves was being reviewed with respect to impact on the

Appendix R safe shutdown analysis. The licensee committed to providing a final update concerning this part of the design within two months of the date of the letter.

No violations or deviations were identified in this area.

3. <u>Licensee Event Reports (LER) Followup (90712 and 92700)</u>

Through direct observations, discussions with licensee personnel, and review of records, the following event reports were reviewed to determine that reportability requirements were fulfilled, immediate corrective action was accomplished, and corrective action to prevent recurrence had been accomplished or planned in accordance with Technical Specifications.

<u>(Closed) LER No. 237/89025</u>: Inadvertent Automatic Isolation of the High Pressure Coolant Injection (HPCI) System Due to Design Deficiency. The activities resulting in this occurrence were discussed in inspection report No. 50-237/89019; No. 50-249/89018. The licensee attributed the root cause of this event to a design deficiency within the Analog Trip System (ATS) panel such that the master trip unit (MTU) mounting configuration can result in spurious trips when adjacent MTUs are removed. The licensee determined that Dresden Instrument Surveillance (DIS) 2300-11, System Isolation-Reactor Pressure Transmitter Calibration and Maintenance Inspection, was the only HPCI instrument surveillance procedure that required removal of adjacent MTUs. Therefore, the licensee planned to incorporate precautions in this procedure to exhibit care when removing and replacing MTUs and to require prior notification to the Operations Shift Supervisor that MTU replacement may result in an isolation signal. The licensee also planned to post signs on the ATS panels to indicate the same caution and requirement. The licensee did not plan to change the MTU mounting configuration since they considered this to be an isolated event and MTU removal was a rare occurrence due to a high reliability of the component.

<u>(Closed) LER No. 237/89026</u>: Start of Standby Gas Treatment System Due to Loose Reactor Building Ventilation System Radiation Monitor Connection. This event including initial licensee actions was described in inspection report No. 50-237/89019; No. 50-249/89018. In addition, the licensee planned to revise DIS 1700-7, Reactor Building Ventilation (RBV) Radiation Monitor Functional Test, to require checking RBV radiation monitors for loose connections and exposed wiring during the surveillance. The licensee also planned to evaluate possible methods to improve instrument department response time to this type of event and to evaluate a generic radiation monitor troubleshooting procedure.

<u>(Closed) LER No. 237/89027</u>: Postulated Low Pressure Coolant Injection (LPCI) Swing Bus Loss Resulting From Diesel Generator Voltage Regulator Failure Due to Design Deficiency. This item and corresponding licensee actions are described in Paragraphs 7.b.1 and 7.c of this report.

(Closed) LER No. 237/89028: Containment Cooling Service Water (CCSW) Pump Suction Bay Water Level Reduction. This event was discussed in inspection report No. 50-237/89019; No. 50-249/89018. As a long term corrective action, the licensee planned to review methods to proceduralize a program that was initiated to measure water level drop across the trash bars. This would contribute to earlier recognize of CCSW suction level bay decreases.

(Closed) LER No. 237/89029: Elevated HPCI Discharge Piping Temperature Due to Reactor Feedwater System Back Leakage. This item and corresponding licensee actions are described in Paragraphs 5.b.4, 5.b.5 and 7.c of this report and report No. 50-237/89023; and No. 50-249/89022.

(Closed) LER No. 237/89030: Reactor Building Fire Wall Degraded By An Unauthorized Penetration Opening Due to Management Deficiency. This item and corresponding short term licensee actions are described in Paragraphs 5.b.8 and 5.c of this report.

(Closed) LER No. 249/89004: HPCI System Declared Inoperable Due to Failed Room Cooler Fan Drive Belts. This item and corresponding licensee actions are described in Paragraph 5.b.3 of this report.

No violations or deviations were identified in this area except as described in Paragraph 5.b.8 of this report.

4. Plant Operations (71707, 71710 and 93702)

a. Enforcement History

During this inspection period, no violations or deviations were identified in the plant operations functional area.

b. Operational Events

On October 10, 1989, the Unit 2/3 Cribhouse Basement Cable Tray Fire Suppression Deluge System was inadvertently actuated during performance of Dresden Fire Protection Procedure (DFPP) 4114-6, Fire System Yard Loop Monthly Inspection, Revision 10. While inspecting the protectowire fire alarm control panel and power supply for the cribhouse basement cable tray fire detection system, fire panel 2223-112, the operator attempted to replace burned out light bulbs as required by the procedure. In order to identify the burned out bulbs, the operator depressed a panel button labeled Alarm Devices-Push to Test, which he thought would just illuminate the panel lights. However, this button instead tested the fire panel relays which actuated the deluge system spraying water into the Unit 2/3 cribhouse basement. The operator immediately isolated flow by breaking the locking device on cribhouse cable tray isolation valve 2/3-4199-176 and closing the valve. A second initiation occurred later that same day due to grounds on the protectowire located in the cable trays which were caused by water from the first initiation. The area was allowed to dry out and inspections revealed no other equipment damage.

c. Approach to the Identification and Resolution of Technical Issues From a Safety Standpoint

The licensee exhibited regard toward ensuring operators were aware of adverse conditions, their affect on the plant and mitigation techniques. This was exemplified by informing operators of an alternate method to determine if Electrohydraulic Control (EHC) DC power were lost as described in Paragraph 5.b.1 of this report. Due to a relay failure at that time, a loss of EHC DC power would have rendered various main turbine trips inoperable without a corresponding alarm to warn the operator of this condition. Questioning of the operators by the inspectors indicated that they were aware of the alternate method.

The licensee's investigation into the inadvertent deluge system actuation represented a thorough and comprehensive root cause analysis and corresponding corrective actions. The licensee attributed the cause to inaccurate labeling which did not make the function of the pushbutton apparent. In addition, DFPP 4114-6 was deficient in that it did not caution the operator concerning this pushbutton. Finally, the licensee determined that operator training was deficient in that the fire system lesson plan also did not provide this information. As a result, the licensee installed an additional label below the pushbutton that read Push to Initiate Deluge. The licensee also proposed the following corrective actions to ensure this event would not be repeated with respect to other fire protection panels:

(1) Discuss the event in Operations and Maintenance tailgate sessions such that personnel are aware of this pushbutton in protectowire fire panels.

(2) Identify all protectowire fire panels that have an equivalent pushbutton and provide the additional warning labels below each of the pushbuttons.

(3) Revise DFPP 4114-6 to identify protectowire fire panels which do not contain a light test button.

(4) Revise the fire system training lesson plan to include this event and to stress the existence of this pushbutton.

(5) Determine the requirements for having the pushbutton in protectowire fire panels and remove those not required.

d. Responsiveness to NRC Concerns

Issuance of Dresden Operating Abnormal (DOA) Procedure 0500-02, Partial Half or Full Scram Actuation, in November 1989 was in response to NRC concerns and indicated the ability to apply lessons learned from other plants. This procedure prescribed mitigating operator actions upon a half or full scram for which Reactor

Protection System scram solenoid indicating lights do not extinguish as they should. This procedure was developed as a result of commitments made to the NRC following such an event at Commonwealth Edison's LaSalle plant.

e. Assurance of Quality, Including Management Involvement and Control

The licensee's decision involving when to initiate a Unit 2 shutdown due to the HPCI piping support damage as discussed in Paragraph 5.b.4 of this report demonstrated management involvement and a desire to ensure that technical specification requirements were met. Previous licensee guidance had concerned the case in which a 24 hour shutdown Limiting Condition for Operation (LCO) was immediately entered. In that case, the licensee's interpretation did not require immediately reducing power if it was legitimately felt that the problem could be rectified and the LCO exited in sufficient time such that an orderly shutdown could still be completed within the original 24 hours if needed. However, the case in question differed from previous guidance in that a seven day LCO was entered prior to entry into the 24 hour shutdown LCO verses being immediately placed into the 24 hour shutdown LCO. Thus, the guidance was unclear as it applied to this situation. To ensure compliance with the requirements, the licensee consulted with NRC regional upper management as to the applicability of previous guidance to this situation. As the licensee felt that actions to consider the system operable could be completed within 12 hours, the decision was made to actually begin the shutdown 12 hours after entry into the 24 hour LCO. This left enough time for completion of an orderly shutdown within the original 24 hours in case the actions did not get completed on time. When the actions were not completed on time, the licensee initiated the shutdown at the time agreed to with the NRC. The inspectors also noted during discussions with licensed operators regarding the incident that they possessed a genuine desire to ensure conservative compliance with technical specifications and, in fact, were concerned as to what appeared to several of them to be actions possibly contrary to previous guidance that they had received in this area. The inspectors regarded this concern to be indicative of a professional attitude of the licensed operators toward their individual licensed responsibilities.

f. Observation of Operations

The inspectors observed control room operations, reviewed applicable logs and conducted discussions with control room operators during this period. The inspectors verified the operability of selected emergency systems, reviewed tagout records and verified proper return to service of affected components. Tours of Units 2 and 3 reactor buildings and turbine buildings were conducted to observe plant equipment conditions, including potential fire hazards, fluid leaks, and excessive vibrations and to verify that maintenance requests had been initiated for equipment in need of maintenance. The inspectors also walked down various HPCI piping supports to ascertain damage and verify repairs as described in Paragraphs 5.b.4 and 5.b.5 of this report.

The inspectors, by observation and direct interview, verified that the physical security plan was being implemented in accordance with the station security plan.

The inspectors reviewed new procedures and changes to procedures that were implemented during the inspection period. The review consisted of a verification for accuracy, correctness, and compliance with regulatory requirements.

The inspectors also witnessed portions of the radioactive waste system controls associated with radwaste shipments and barreling.

These reviews and observations were conducted to verify that facility operations were in conformance with the requirements established under Technical Specifications, 10 CFR, and administrative procedures.

5. Maintenance and Surveillance (62703, 61726 and 93702)

a. Enforcement History

During this inspection period, one violation was identified in the maintenance/surveillance functional area. This concerned a failure to properly control the design of a penetration through a fire barrier such that maintenance personnel degraded that barrier on two separate occasions.

b. Operational Events

Various maintenance activities associated with the following events were observed or reviewed to ascertain that they were conducted in accordance with approved procedures, regulatory guides and industry codes or standards and in conformance with technical specifications.

The following items were considered during this review:

The LCOs were met while components or systems were removed from service; approvals were obtained prior to initiating the work; activities were accomplished using approved procedures and were inspected as applicable; functional testing and/or calibrations were performed prior to returning components or systems to service; quality control records were maintained; activities were accomplished by qualified personnel; parts and materials used were properly certified; radiological controls were implemented; and, fire prevention controls were implemented. Work requests were reviewed to determine status of outstanding jobs and to assure that priority is assigned to safety related equipment maintenance which may affect system performance.

(1) On October 12, 1989, alarms for EHC DC Power Failure and EHC Electrical Malfunction were received on Unit 3. This was of particular concern since loss of EHC DC power would render many of the main turbine trips inoperable. Troubleshooting activities conducted by instrument maintenance and witnessed by the inspectors indicated that DC power was still available and that the alarm relay itself was malfunctioning. However, it was decided not to replace the relay since such an action would be highly susceptible to causing a main turbine trip. The relay in question was located on a circuit card which also contained several other trip relays. These relays were of a mercury type such that inappropriate movement when replacing the card could cause a trip. Thus, the licensee intended to wait until the next time power was reduced to less than 45% to repair the problem so that a turbine trip would not also result in a reactor scram.

(2) On October 15, 1989, the breaker for the Unit 2 LPCI Room Cooler B was found to have been damaged when operators investigated a report that smoke was seen coming from the breaker. The inoperability of the room cooler also required Core Spray Loop B and LPCI Loop B to be declared inoperable, placing Unit 2 into a 24 hour required shutdown LCO. The breaker was, however, repaired later that same day such that the shutdown did not have to commence.

(3) On October 22, 1989, the Unit 3 HPCI System was declared inoperable due to discovery of broken fan belts on the HPCI room cooler. This placed the unit into a seven day LCO. The belts were replaced and the system declared operable on October 23, 1989. The licensee had previously planned to take the room cooler out-of-service on October 23, 1989, for bearing work. Thus, this activity was also completed. A previous event concerning Unit 2 HPCI room cooler broken fan belts was discussed in inspection report No. 50-237/89019; No. 50-249/89018. The root cause of that event was determined to be excessive use of the room cooler due to elevated HPCI room temperatures caused by feedwater system backleakage into the feedwater lines. Increased HPCI line temperatures eventually led to inoperability of the Unit 2 HPCI system as discussed in Paragraph 5.b.4 of this report. However, the licensee indicated that the Unit 3 HPCI room and the HPCI line temperatures were much less than on Unit 2. Thus, the licensee initially indicated that these events were unrelated and backleakage was not a problem on Unit 3 HPCI.

The licensee attributed the cause of the Unit 3 HPCI room cooler belt failure to be shaft misalignment due to the worn bearing. Although the exact cause of the worn bearing was unknown, the most probable cause was inappropriate drive belt tensioning. Dresden Electrical Procedure (DEP) 5700-4, Electrical Maintenance and Surveillance of HPCI Room Fan Motors,

instructed the user to ensure proper belt tension was achieved but gave no additional guidance as to what this tension should be. Therefore, the licensee planned to revise DEP 5700-4 to include proper belt tension information.

(4) On October 23, 1989, the licensee found the Unit 2 HPCI system discharge piping water temperature to be sufficiently high to potentially cause voids to form within the piping. Piping temperatures were discovered to have increased to 275 degrees F between HPCI pump discharge outboard valve 2-2301-8 and HPCI pump discharge inboard valve 2-2301-9, 246 degrees F at the HPCI pump and 135 degrees F near the condensate storage tank (CST). The corresponding static pressure in the HPCI discharge piping at the pump was 32 psig (47 psia). Thus, temperature and pressure in particular areas of the system, represented possible saturated conditions which the licensee believed provided the potential for a waterhammer event. Therefore, the licensee declared the Unit 2 HPCI system inoperable and entered a seven day LCO. On October 27-28, 1989, the licensee discovered numerous signs of damage to various Unit 2 HPCI discharge piping supports. An unusual event (UE) was declared on October 31, 1989, when the licensee initiated a technical specification required shutdown due to a failure to return the HPCI system to operability within the seven day LCO. The system was returned to operability that same day prior to completion of the shutdown. This event, including licensee corrective actions, was discussed in detail in inspection report No. 50-237/89023; No. 50-249/89022. A clamp on Unit 2 HPCI piping support M-1151D-154 located on top of the torus was identified to be rotated on the pipe and a work request initiated during the last Unit 2 refueling outage. However, this work request was not completed during that outage. This is considered part of an unresolved item (No. 237/89022-01(DRP)), together with the item in Paragraph 5.b.5 of this report, pending NRC review and determination of why this work was deferred.

(5) On October 29, 1989, the licensee found the Unit 3 HPCI system discharge piping temperature at an elbow of the piping near its emergence from the X-area (steam tunnel) to be 256 degrees F. Additional measurements obtained on October 31, 1989, indicated piping temperature just upstream on the other side of the elbow measured between 163 and 133 degrees F depending on the circumference location. The corresponding static pressure in the HPCI discharge piping at the pump was about 45 psig (60 psia). The licensee believed temperature and pressure conditions near the elbow could potentially cause steam pocket formation. Thus, the licensee declared the Unit 3 HPCI system inoperable. On November 1, 1989, the licensee also discovered signs of damage to Unit 3 HPCI piping supports. The Unit 3 HPCI system was returned to service on November 7, 1989. This event including

licensee corrective actions, was discussed in detail in inspection report No. 50-237/89023; No. 50-249/89022. Unit 3 HPCI piping support M-1187D-110 was found to have the baseplate and all four concrete expansion anchors pulled from the wall. Evidence also showed that a licensee walkdown conducted in 1979 noted a wallmount pulling away. This is considered part of an unresolved item (No. 237/89022-01(DRP)), together with the item in Paragraph 5.b.4 of this report, pending NRC determination of whether this is the same damage as originally identified.

(6) On November 6, 1989, the Unit 2 HPCI Motor Gear Unit (MGU) high speed stop (HSS) indicating light was discovered to be blinking on and off. However, the MGU was still functional since it automatically returned to it's HSS from it's low speed stop (LSS). A large amount of noise was discovered in the DC output signal and, thus, the HPCI MGU was taken out of service to repair it on November 8, 1989. The MGU HSS indication fluctuations were eliminated by replacement of a circuit capacitor and HPCI was declared operable on November 10, 1989.

(7) Throughout much of the inspection period, Unit 2 operated at slightly reduced power due to repeated spurious primary containment half isolation signals received at full power conditions. These half isolations were caused by failure of main steamline low pressure switch PS-261-30B. The licensee believed that rapid pressure fluctuations within the pressure line caused by vibration was prematurely degrading the bourdon tube within the switch. This had been a recurring problem in the past with previous actions involving vibration testing of the main steamline low pressure switches and installation of a pressure snubber in the sensing line. The switch had been replaced several times but would typically fail after approximately one month. Load was reduced to 65 percent on November 18, 1989, in order to allow entry to the heater bay to conduct a walkdown of the sensing line. This walkdown did not identify any problems with the line. On November 22, 1989, PS-261-30B was replaced and a new portion of sensing line on the instrument rack was installed in a looped configuration in hopes of dampening any pressure fluctuations to the switch. The licensee was also evaluating possible future replacement with a different and less susceptible type switch.

(8) On October 26, 1989, the Station Manager discovered a three inch open penetration stuffed with rags in a three hour fire rated wall separating the Units 2 and 3 reactor buildings at elevation 570 feet. The mechanical maintenance department was in the process of dismantling and cleaning an area on the Unit 2 side of the wall which was formerly a control rod drive (CRD) maintenance area. The work being performed under a blanket work request for general plant cleanup was not intended to disrupt or alter plant components or systems. A drain line connected to a CRD flush tank had previously been routed

through the penetration to a floor drain on the opposite side of the wall. Due to high radiation levels from the drain line and the fact that the CRD flush tank was to be removed during the cleanup, removal of the drain line was also added to the scope of the work. Maintenance personnel did not realize that the wall was a rated fire barrier or that it would be degraded by the open penetration, although a nearby fire door in the same wall was present and easily identifiable. Under a normal work request, a determination by the working department would have been required as to whether a fire hazard review by the fire marshall should be accomplished during the work planning stage. This would have included a review to determine the applicability of DFPP 4175-1, Fire Barrier Integrity and Maintenance, and DFPP 4175-2, Operating Fire Stop/Break Surveillance. However, a blanket work request bypassed these types of controls. Approximately 24 hours elapsed between the time the piping was removed from the penetration and discovery by the Station Manager. During this period of time, an hourly fire watch, although required as a result of the inoperable penetration by Dresden Administrative Technical Requirement (DATR) 3.1.6.1, did not exist. The DATRs were first implemented on August 29, 1989, to incorporate fire protection requirements that were deleted from technical specifications as described in Paragraph 7.b.2 of this report.

A previous event also involving degradation of a fire barrier by maintenance personnel occurred on June 14, 1989. Failing to recognize a fire barrier, workers routed a welding cable and air hose through an unducted ventilation opening in the fire wall separating the Unit 3 east LPCI room and the Unit 3 HPCI room. This prevented closure of an automatic vertical fire damper in the ventilation opening. The technical specification requirement in effect at that time required a continuous fire watch to be established within one hour due to the inoperable fire barrier penetration. This was not established until the degradation was discovered three days after it occurred. This event was described in inspection report No. 50-237/89017; No. 50-249/89016. NRC review indicated that this previous event met the criteria of 10 CFR 2, Appendix C, and thus no notice of violation was issued at that time. Corrective action to prevent recurrence involved marking of unducted ventilation openings in fire barriers to make them more recognizable and, therefore, was very specific to that event. This corrective action also was not complete at the time of this latest occurrence in that of five identified unducted ventilation openings in fire barriers only one had already been appropriately marked. The remaining were to be completed during the December 1989 Unit 3 refueling outage. This action did not address the broader aspects of maintenance personnel recognition of fire barriers in general and, therefore, could not have prevented this latest occurrence even if it had been completed.

Further review by the licensee determined that the rated fire assembly penetration had been degraded even prior to the piping removal. The penetration was originally installed in 1982. However, at some date between April 1, 1985 and July 8, 1985 sections of the piping including the portion going through the penetration were replaced with polyvinyl chloride (PVC) plastic piping, a non-approved material for fire barrier penetrations. Plastic materials will burn with an intensity and heat production in a range similar to that of ordinary hydrocarbons. In addition, when burning, they produce heavy smoke that obscures visibility and can plug air filters. The halogenated plastics also release free chlorine and hydrogen chloride when burning, which are toxic to humans and corrosive to equipment. The work request under which this change was completed indicated that no fire hazards review was necessary.

The design drawing, fire barrier location drawing F-88, failed to identify the penetration. Drawing F-88 was inspected by the architect-engineer (AE) for fire barrier drawing development on February 14, 1985. This inspection was to identify all penetrations in the fire wall including both mechanical and electrical penetrations.

In addition, performance of surveillance DFPP 4175-2 failed to identify the existence of the penetration. This surveillance, required to be performed on an 18 month cycle, contained specific instructions to enter data on the Operating Fire Stop Surveillance Log and initiate a drawing change request for the appropriate fire protection drawing if a fire barrier penetration was found that was not on the drawings. Instructions for review of mechanical penetration seals were incorporated into the procedure on December 29, 1986 with Revision 5 of the procedure. Previous revisions required inspection only with respect to electrical fire seal penetration configurations. Inspections per this procedure including those pertaining to mechanical penetration seals were accomplished on February 1, 1988 and again on February 1, 1989, each time failing to identify the penetration in question.

This is considered to be a violation of 10 CFR 50.48(a) (No. 237/89022-02(DRP)) in that the licensee failed to control the design for this fire rated assembly (fire wall). The penetration was not identified during Appendix R walkdowns, was not included on fire protection drawings, and was not identified through the fire protection surveillances on the fire barrier. Furthermore, the fire rated wall was degraded in 1985 by installation of combustible PVC piping and again recently with complete removal of the piping. Each time, the effect on the fire barrier was not properly analyzed or considered. The cause of the more recent degradation of the fire barrier was, in fact, similar in nature to a fire barrier degradation which occurred earlier this year. In both, maintenance personnel failed to recognize a fire barrier and, therefore, the effect their actions would have on it.

c. Approach to the Identification and Resolution of Technical Issues From a Safety Standpoint

The licensee's approach to resolution of technical issues in the maintenance area was mixed as demonstrated by the violation associated with the fire barrier degradation as opposed to the actions associated with the EHC DC power failure alarm relay.

Licensee corrective actions to the June 1989 fire barrier degradation by maintenance personnel, in retrospect, proved to be too narrow in scope to prevent another fire barrier degradation. Upon discovery of the later degraded fire barrier described in Paragraph 5.b.8 of this report, the licensee initiated an hourly fire watch. A temporary fire seal was installed on October 26, 1989, and a permanent seal was installed on November 17, 1989, when proper materials were available. The decision to wait for better conditions prior to replacing the EHC DC power failure alarm relay as described in Paragraph 5.b.1 of this report was an example of a regard for minimization of unplanned transients. In this way, if a main turbine trip would result from the activity, it would not also cause a reactor scram. The inspectors also noted that instrument maintenance personnel troubleshooting the problem were highly knowledgeable of detailed EHC system circuitry design. Licensee actions taken in response to the main steamline low pressure switch failures was regarded by the inspectors to be a good attempt to identify the specific problem and resolve it.

d. Responsiveness to NRC Initiatives

The licensee's timeliness of control room work request completions continued to be in response to NRC concerns. To ensure prompt resolution of such problems the licensee revised Operations Department Policy Statement Number 16. This statement established a white work request sticker for the control room to be used in addition to the existing salmon colored stickers. A salmon sticker was to be used to identify problems with control room indications such that the operator could no longer believe the indication or the indication was no longer available. A white sticker was used to identify problems that required corrective maintenance but control room indications were not affected. Salmon stickers were to receive a B-1 priority which required work to start within 24 hours if parts were available.

e. Observation of Surveillance Activities

The inspectors observed surveillance testing required by Technical Specifications for the items listed below and verified that testing was performed in accordance with adequate procedures, that test instrumentation was calibrated, that LCOs were met, that removal and restoration of the affected components were accomplished, that test results conformed with Technical Specifications and procedure requirements and were reviewed by personnel other than the individual directing the test, and that any deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel.

The inspectors witnessed portions of the following test activities pertaining to Units 2 and/or 3:

Local Power Range Monitor (LPRM) Amplifier Gain Calibration
Average Power Range Monitor (APRM) Gain Adjustment
Individual LPRM Recovery
Quarterly Primary Containment Isolation Valve Timing
APRM Rod Block and Scram Functional Test
Intermediate Range Monitor Downscale Rod Block Functional Test
HPCI Valve Operability Test

6. <u>Safety Assessment/Quality Verification (40500)</u>

a. Enforcement History

During this inspection period, no violations or deviations were identified in the safety assessment/quality verification functional area.

b. Assurance of Quality, Including Management Involvement and Control

Management involvement in assuring quality was evident when the plant manager discovered the degraded fire wall as described in Paragraph 5.b.8 of this report. The inspectors continued to note frequent and effective tours of the plant by management.

The inspectors observed the monthly performance review meeting conducted on October 13, 1989. Plant management reviewed items of interest which occurred since the last meeting including engineered safety feature actuations, specific Technical Specification limiting conditions for operation entered, continuous or occurring control room alarms, degraded or out of service equipment and potentially significant events. In addition, the status of the top technical issues was discussed. In order to facilitate greater sharing of information with similar facilities, a representative from the Quad Cities plant was also present. In addition, the meeting was attended by a licensed plant operator who presented his own areas of concern. The inspectors considered attendance by both these individuals to be beneficial toward maintaining management awareness and involvement in relevant issues both internal and external to the plant. Attendance by plant operators also tended to promote greater professionalism and a sense of responsibility among that group.

The inspectors also reviewed the monthly status report for the month of October. The inspectors found this to be an excellent management tool for remaining cognizant and identifying trends in various departmental indicators.

7. Engineering/Technical Support (37828 and 93702)

a. Enforcement History

During this inspection period, no violations or deviations were identified in the engineering/technical support functional area.

b. Operational Events

(1) The licensee informed the resident inspectors on October 12, 1989, that they had confirmed a possible single failure that could occur during a loss of coolant accident (LOCA) following a loss of offsite power that could prevent the low pressure coolant injection (LPCI) swing bus, MCC 28-7/29-7 on Unit 2 (MCC 38-7/39-7 on Unit 3), from performing its intended function. The LPCI swing bus could be supplied power from either bus 28 or bus 29 on Unit 2 (bus 38 or bus 39 on Unit 3) which in turn were supplied power from opposite engineered safety feature (ESF) divisional buses. A low voltage condition on the LPCI swing bus was designed to cause an automatic transfer of the bus to the bus supplied from the other division. However, a diesel generator could suffer a voltage regulator failure such that voltage would be too low to properly operate bus loads but not low enough to cause the LPCI swing bus to automatically transfer to the division supplied by the other diesel generator. The LPCI injection valves were supplied power from the LPCI swing bus. Thus, the LPCI system and one division of core spray would be incapable of automatic injection in this scenario. This would leave only one core spray pump for automatic low pressure emergency core cooling system (ECCS) injection.

(2) On November 16, 1989, the licensee discovered that a DATR involving a fire detection instrument had been inadvertently missed. Technical Specification amendment numbers 106 for Unit 2 and 101 for Unit 3 removed the fire protection requirements from technical specifications in accordance with guidance presented in Generic Letters 86-10 and 88-12. The DATRs incorporated these technical specification requirements while also including the fire protection features added during the 10 CFR 50 Appendix R fire protection modifications. This included the addition of LCO actions to reflect the added fire protection features. These technical specification amendments were approved by the NRC on June 29, 1989, with 60 days given to implement the change. In preparation for implementation, work requests were reviewed by the system engineer and the fire marshall to see if inoperable equipment was affected by the DATRs. A total of 26 work requests were identified including one involving the Unit 3 LPCI room/torus fire detection (protectowire) device which was written on July 26, 1989. The associated DATR 3.I.1.1 LCO action statement required a once per hour fire inspection to be established within one hour. However, the work request review inappropriately identified

another action statement which was applicable to the other work requests as also applicable to this work request. This other action statement allowed 14 days to restore the device prior to establishing the fire watch. Thus, when the DATRs became effective on August 29, 1989, the fire watch was not established. On September 12, 1989, when the 14 days expired, the fire watch was established and a deviation report written. The device was repaired and considered operable on September 23, 1989. While reviewing the deviation report on November 16, 1989, the system engineer discovered the error.

The inspectors regarded this incident as an isolated occurrence induced by implementation of the new program requirements and a review process which differed from normal practices. The inspectors had not noted any further problems with DATR compliance under normal practices since their implementation, except as described in Paragraph 5.b.8 of this report. This exception, however, was attributable to a different root cause.

(3) While assembling work packages to install and calibrate United Electric Temperature switches for main steamline and HPCI steamline leak detection and automatic isolation, the licensee discovered that the model F100 switches to be installed were not referenced in the environmental qualification (EQ) binder. Further review by the licensee on November 14, 1989, indicated that five of the 16 Unit 2 main steamline temperature switches were already installed without the proper EQ documentation. One of these was installed in February 1989 and the other four in July 1989. The other Unit 2 main steamline, as well as all Unit 3 main steamline and Units 2 and 3 HPCI steamline temperature switches were properly EQ qualified model F7 switches. Although the suitability of application previously completed by the licensee for the F100 switch indicated that it was EQ qualified, this determination was based on a vendor test report and not on the required EQ binder. This is considered to be an unresolved item (No. 237/89022-03(DRP)) pending further NRC review of this matter.

c. Approach to The Identification and Resolution of Technical Issues From a Safety Standpoint

The licensee's determination of the LPCI swing bus design problem indicated a commitment toward remaining cognizant of industry issues and problems that could be relevant to Dresden. The review that identified this problem was implemented in response to similar deficiencies discovered at other nuclear power plants. Licensee subsequent actions included evaluating possible design changes and contacting the facilities with similar identified deficiencies to ascertain their respective courses of action. Two possibilities that were under review included additional protective relays or powering the involved motor control centers with an uninterrupted power supply. The licensee also issued Dresden General Abnormal

(DGA) Procedure 5, Degraded Voltage on MCC 29-7/28-7 (39-7/38-7) Due to a Failure of the Unit 2(3) Diesel Generator Voltage Regulator During a LOCA/Loss of Offsite Power Event. This procedure required the operator to trip the diesel generator if adequate voltage could not be restored such that the LPCI swing bus would automatically transfer. If this attempt failed, the operator was instructed to manually transfer the LPCI swing bus.

The inspectors regarded the missed DATR concerning the fire protection protectowire device to be an excellent example of a commitment to self-identification of problems by not only the licensee but also the individual who discovered and reported his own error. The licensee planned to include a discussion of the incident in station personnel tailgate sessions and in the licensed operator requalification continuing training program. The licensee also identified the EQ problem regarding five of the Unit 2 main steamline temperature switches. As a result, the licensee completed equipment qualification variation form 89-023 including a justification for continued operation. An EQ binder was also being developed to rectify the problem.

The inspectors regarded the licensee investigation, root cause analysis and corrective actions concerning the HPCI system backleakage and damaged piping supports, as described in Paragraphs 5.b.4 and 5.b.5 of this report as an example of aggressive self identification and resolution of problems. The review of elevated room temperatures and corresponding actions which led to discovery of the feedwater backleakage into the HPCI system was particularly insightful. The system walkdowns used to identify the HPCI support damage were very detailed and comprehensive. In addition, safety evaluations performed to support alternate HPCI system standby lineups addressed all relevant issues. Planned licensee actions to determine the root cause of HPCI system valve leakage, to access the effectiveness of the Inservice Inspection (ISI) program as it applied to structural supports and to perform similar walkdowns on other systems indicated an excellent attitude toward self-identification and assessment.

d. Responsiveness to NRC Concerns

The plant technical staff was responsive to a regional NRC request for information regarding maintenance of shutdown margin requirements during refueling.

8. Report Review (90713)

During the inspection period, the inspectors reviewed the licensee's Monthly Operating Report for October. The inspectors confirmed that the information provided met the requirements of Technical Specification 6.6.A.3 and Regulatory Guide 1.16. The inspectors also reviewed the Unit 2 Cycle 12 Startup Test Report Summary and confirmed that it met the requirements of Technical Specification 6.6.A.1.

9. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, violations or deviations. Unresolved items disclosed during the inspection are discussed in Paragraphs 5.b.4, 5.b.5 and 7.b.3 of this report.

10. Exit Interview (30703)

The inspectors met with licensee representatives (denoted in Paragraph 1) on December 1, 1989, 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 inspector during the inspection. The licensee did not identify any such documents/processes as proprietary. The licensee acknowledged the findings of the inspection.

Commonwealth Edison
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January 25, 1990

Mr. A. Bert Davis
Regional Administrator
U.S. Nuclear Regulatory Commission
Region III
799 Roosevelt Road
Glen Ellyn, IL 60137

Subject: Dresden Nuclear Power Station Units 2 and 3
Response to Notice of Violation and Inspection
Report Nos. 50-237/89022 and 50-249/89021
NRC Docket Nos. 50-237 and 50-249

Reference: Letter from W.D. Shafer to Cordell Reed dated
December 26, 1989, transmitting the subject Inspection
Report and Notice of Violation.

Mr. Davis:

Enclosed is the Commonwealth Edison Company (CECo) response to the subject Notice of Violation (NOV) and Inspection Report (IR) which identified deficiencies in the control of a fire barrier penetration.

CECo understands the significance of the issues involved. Corrective actions have been taken or have been initiated to prevent similar non-compliances from recurring in the future.

Please contact this office should further information be required.

Very truly yours,

T.J. Kovach
Nuclear Licensing Manager

cc: B.L. Siegel - Project Manager, NRR
S.G. DuPont - Senior Resident Inspector, Dresden

lw/0593T

ATTACHMENT
COMMONWEALTH EDISON COMPANY

Response to Notice of Violation 50-237/89022-02 (DRP)

Severity Level IV

VIOLATION

10CFR50.48(a) requires that each operating nuclear power plant have a fire protection plan that satisfies Criterion 3 of Appendix A to 10CFR Part 50. It further requires that the plan shall describe specific features necessary to implement the plan such as administrative controls to limit fire damage to structures, systems or components important to safety so that the capability to safely shutdown the plant is assured.

Section C.1 of the licensee's response to the Guidelines of Appendix A to Branch Technical Position APCSB 9.5-1 as accepted in the 1980 Supplemental Safety Evaluation Report indicates that administrative measures are established to ensure that guidelines of the Branch Technical Position are included in design and procurement documents and that deviations therefrom are controlled.

Contrary to the above, a penetration in a three hour fire rated wall located in a safety related area of the 570 foot elevation of the reactor building, as prescribed by Section D.1.j of the Branch Technical Position, was not included in design documents and deviations were not controlled. The fire rated wall was degraded in 1985 by replacement of the original piping with non-approved polyvinyl chloride plastic piping and was further degraded on October 25, 1989 when the piping was completely removed and the penetration left unsealed.

This is a Severity Level IV violation (Supplement I) [No. 237/89022-02 (DRP)].

DISCUSSION

The Station's Technical Specifications include a license amendment that requires adherence to the approved fire protection program. This amendment is implemented through the Dresden Administrative Technical Requirements (DATRs) for fire protection. DATR 3.1.6.1.a requires that a fire watch be posted within one hour whenever a fire rated sealing device is inoperable. Because the investigation into this event established that the fire watch time constraint was exceeded, this event was reported under 10CFR50.73(a)(2)(i)(B) for a condition that is prohibited by the Technical Specifications (LER 89-30/050237).

The Mechanical Maintenance Department was in the process of dismantling and cleaning an area which was formerly a Control Rod Drive (CRD) maintenance area. This work was being performed under Blanket Work Request No. 208 for general plant cleanup. The work that was to be performed was not intended to disrupt or alter plant components or systems. The work described on the

Blanket Work Request form must be approved by a Maintenance Department Supervisor. Prior to commencing on the cleanup work, the Radiation Protection Department surveyed the work area and identified a drain line as a source of high radiation. The drain line was connected to a CRD flush tank and routed through the Unit 2 and Unit 3 Reactor building common wall directly to a floor drain. Because the CRD flush tank was to be removed per the blanket work request, removal of the drain line was improperly added to the blanket work request job scope. In order to reduce personnel exposure, the drain line was hydrolazed and removed before other work in the area resumed.

A substantial portion of the line was hydrolazed and removed between October 24 and October 26, 1989. On October 26, 1989, the final portion of pipe remaining in the Reactor Building common wall was removed. At approximately 1100 hours, the Maintenance Mechanics stuffed the penetration with rags and left the area.

Further investigation into this incident revealed that the drain line penetration was originally installed in 1982 per fire protection requirements for a three hour barrier. However, subsequent to its initial installation, sections of the piping were replaced with Polyvinyl Chloride (PVC) plastic piping including the portion that went through the common wall penetration. Further investigation revealed that the PVC pipe alteration occurred in 1985 when it was insufficiently described in the associated Work Request to be identified as involving a fire barrier penetration. The scope of the Work Request as written was to improve the drainline flow by changing the angularity of the pipe. Consequently, it was not identified as Reliability or Regulatory Related. Since that time, the quality of work instructions has been upgraded and all fire protection related work is classified as Regulatory Related which requires review by the Fire Marshall as well as Quality Control.

The most recently performed Technical Staff Fire Protection Procedure (DFPP) 4175-2, "Operating Fire Stop/Break Surveillance," failed to identify the drain line penetration. The fire barrier location drawings, which were first issued in 1985 following a detailed fire barrier survey, also failed to show the penetration. The DFPP 4175-2 surveillance, which is performed on an 18 month cycle, includes instructions to inspect Appendix R wall and floor fire barriers for evidence of new penetrations or breaches. If an unrated penetration seal or breach in an Appendix R fire barrier is identified, the Operations Department Shift Supervisor is to be notified to implement immediate corrective actions. The penetration would then be documented in the surveillance procedure, and in the fire barrier location drawings. It is believed that performance of the penetration surveillance was hampered due to the continuing maintenance work in the areas on either side of the wall. The surveillance technicians's line of sight was most likely obscured or obstructed in each case while inspecting the third floor Unit 2/3 Reactor Building wall, thus preventing detection of the drain line penetration.

<u>CORRECTIVE ACTIONS TAKEN AND RESULTS ACHIEVED</u>

The immediate corrective actions were notification of Operations Department Shift Supervision, and the initiation of an hourly fire watch pursuant to DATR 3/4.1.6. The penetration was then sealed with a temporary fire seal in accordance with Dresden Fire Protection Procedure (DFPP) 4175-1, "Fire Barrier Integrity and Maintenance." Once the temporary fire seal was inspected and approved, the fire watch was terminated. Contrary to DFPP 4175-1, however, a

permanent seal was not installed within the prescribed seven days. Materials to make the repair were not available in time to complete the repair. The Station Fire Marshall, at his discretion, permitted the seven day administrative limit to expire provided that the temporary barrier was intact, and that the permanent barrier was installed as soon as practicable. Mechanical Maintenance installed the permanent seal under Work Request 88289 on November 17, 1989.

CORRECTIVE ACTIONS TAKEN TO AVOID FURTHER NON-COMPLIANCES

1. DFPP 4175-2, will be revised by the Technical Staff to include this fire seal on the surveillance checklist. Also, to aid in performing the next fire barrier surveillance, a Drawing Change Request (DCR) will be initiated to identify the fire seal location on fire barrier drawings F-88 sheets 1 and 2. This will be completed by February 28, 1990.

2. In order to make rated fire walls in the plant more easily identifiable, the Technical Staff system engineer will prepare a fire barrier reference guide including plan views of all the fire areas for use by all working departments. A revision to Dresden Administrative Procedure (DAP) 3-1, "Fire Protection Program," will also be implemented to control preparation and updating of the reference guide. This will be completed by July 31, 1990.

3. The Fire Marshall will provide the Training Department with additional training material on fire barriers by February 12, 1990.

4. Additional training on fire barriers will be given to the Mechanical Maintenance Department during an upcoming continuing training session. A review of this event shall be included in the material to be presented. Emphasis will be placed on the conservative practice of assuming that all walls, floors, and ceilings in the Reactor and Turbine Buildings are fire barriers unless otherwise specified. This will be completed by May 25, 1990.

5. This event was reviewed in a tailgate meeting for all station personnel on December 21, 1989. The conservative practice described in Item 4 will also be emphasized in additional tailgate meetings for all station work groups, substation construction, and ENC to be completed by February 23, 1990. It will be included in entrance training for contractor personnel by May 25, 1990.

6. This event will be reviewed with the Mechanical Maintenance Supervisor and Crew who were directly involved by January 31, 1990.

7. A statement on the appropriate use of the Blanket Work Request system was added to DAP 15-1 by the Maintenance Staff on January 12, 1990.

8. Precautionary statements will be added to fire barrier surveillance procedures DFPP 4175-2 and DFPP 4175-3 ("Shutdown Fire Stop/Break Surveillance") concerning:

   a) improperly modified penetrations, and
   b) removal of obstructions, as appropriate, in order to assure that the entire barrier is properly inspected.

   These procedure changes will be implemented by June 29, 1990, i.e. prior to the next 18 month surveillance.

9. Changes have also been implemented in DFPP 4175-1 to clarify the process by which temporary seals may be approved for longer than seven days. It now provides more detailed installation instructions and inspection frequency requirements to ensure that temporary fire seals provide adequate barrier protection for periods exceeding seven days. This was completed on January 12, 1990.

DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED

As described previously, a fire watch was promptly established upon discovery of the degraded fire barrier. The fire barrier penetration opening was then sealed with an approved temporary configuration in accordance with DFPP 4175-1. Once the temporary fire seal was inspected satisfactorily, the fire watch was terminated. Mechanical Maintenance then installed a permanent seal under Work Request 88289. The permanent seal was then inspected satisfactorily on November 17, 1989, at which time all actions to achieve full compliance were complete.