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Commonwealth Edison Company
Dresden Generating Station
6500 North Dresden Road
Morris, IL 60450
Tel 815-942-2920

ComEd

November 21, 1995

JSP Ltr. 95-0020

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Dresden Nuclear Power Station Units 2 and 3
Change of Commitment for Submittal of Proposed License Amendment
NRC Docket Nos. 50-237 and 50-249

References: 1) J. A. Zwolinski to M. J. Wallace letter dated July 12, 1993, Transmitting Notice of Violation, Inspection Report 50-237\92034; 50-249\92034

2) M. D. Lyster letter to NRC Document Control Desk dated September 3, 1993, Transmitting Response to Notice of Violation Inspection Report 50-237/92034; 50-249/92034

In Reference 1), the NRC cited ComEd with a Level III violation for making changes to the Containment Cooling Service Water (CCSW) System with an inadequate 10 CFR 50.59 Safety Evaluation that contained an unreviewed safety question. In Reference 2), ComEd committed that a license amendment request would be submitted to amend the licensing basis for the CCSW System at Dresden Station in order to clarify the design basis of the CCSW System. This license amendment would require a single CCSW pump for containment cooling purposes. Additional commitments included improvements to the CCSW Design Basis Document (DBD) Program and Technical Specification Upgrade Program (TSUP).

ComEd considers enhancements to its DBD Program and TSUP sufficient to clarify the design basis of the CCSW System. Therefore, a license amendment is not required to clarify the design basis and will not be submitted. As such, the licensing basis for the CCSW system will continue to require two (2) CCSW pumps for containment cooling purposes. Any future licensing actions regarding the Containment Cooling Service Water System will be evaluated and discussed with the NRC staff at the time of submittal.

If your staff has any questions concerning this letter, please refer them to Mr. Bohdan Rybak, Nuclear Licensing Administrator, at (708) 663-7292.

Sincerely,

J. Stephen Perry
Vice President
BWR Operations

280038

cc: J. B. Martin, Regional Administrator, Region III
J. F. Stang, Project Manager, NRR
C. Vanderniet, Senior Resident Inspector, Dresden
B. Rybak, Nuclear Licensing Administrator, Downers Grove

A001 1/0

Commonwealth Edison Company
1400 Opus Place
Downers Grove, IL 60515

ComEd

November 14, 1995

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Dresden Nuclear Power Station Units 2 and 3
Quad Cities Nuclear Power Station Units 1 and 2
Supplement to Application for Amendment to Facility Operating Licenses DPR-19, DPR-25, DPR-29 and DPR-30, Appendix A, Technical Specifications for the Technical Specifications Upgrade Program (TSUP)
Clean-Up Package
NRC Docket Nos. 50-237/249 and 50-254/265

References: (see attached)

The purpose of this letter is to supplement various sections of the TSUP project that implement minor changes to previously submitted TSUP packages (see attached References). A summary and ComEd's assessment of the proposed changes are provided as Attachment A to this letter. Attachment B highlights the proposed changes and includes marked-up versions of the affected TSUP pages. Attachment C provides revised TSUP pages reflecting the marked-up changes noted in Attachment B. The evaluation of significant hazards considerations for the proposed TSUP clean-up changes proposed within Reference (37) encompasses the changes proposed herein. For completeness, a supplemental evaluation of significant hazards has been provided as Attachment D.

The proposed changes serve to close-out all open items identified during the NRC staff's review as noted in previous NRC staff Safety Evaluations received for previously provided submittals regarding the TSUP project.

The proposed supplemental changes have been approved by Commonwealth Edison's (ComEd) On-Site and Off-Site Review in accordance with Company procedures. Commonwealth Edison requests that the proposed changes be approved as submitted to become effective upon completion of the entire TSUP project.

To the best of my knowledge and belief, the statements contained in this document are true and correct. In some respects these statements are not based on my personal knowledge, but on information furnished by other ComEd employees, contractor employees, and/or consultants. Such information has been reviewed in accordance with company practice, and I believe it to be reliable.

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As noted in Attachment A, Item 6 of ComEd's submittal regarding the close-out of certain TSUP open items (ComEd's submittal to the NRC staff, dated September 15, 1995), ComEd discussed the basis for acceptance of the Standby Liquid Control System (SBLC) pump surveillance frequency. Within this discussion, ComEd inadvertently specified "Reference (e)" as the originating document. "Reference (e)" included ComEd's submittal for TSUP Section 3/4.10, dated February 16, 1993. The appropriate cross-reference is ComEd's submittal for TSUP Section 3/4.4, dated October 15, 1992. ComEd apologizes for any inconvenience this discrepancy may have caused the NRC staff.

Based upon discussions with members of the NRC staff, ComEd was requested to provide further justification regarding the relocation of TSUP 3/4.10.F to administrative controls. As discussed in ComEd's response to the NRC staff's request for additional information (RAI) for TSUP 3/4.10, dated May 2, 1995), ComEd stated that TSUP 3/4.10.F for both Dresden and Quad Cities (based on current Technical Specifications [CTS] 3/4.10.F) will be relocated to administrative controls. These requirements are relocated to plant controlled documents which is consistent with the Improved Standard Technical Specifications (ITS - NUREG-1433) and does not adversely affect existing plant "heavy loads analyses" for Dresden or Quad Cities Stations. These requirements shall continue to be enforced but will be administratively controlled per the provisions of 10 CFR 50.59 at both Dresden and Quad Cities Station and will prohibit improper loads from being transported at the sites. As such, the proposed TSUP package does not reduce existing safety margins, does not adversely affect the current licensing basis, does not adversely affect Dresden or Quad Cities "heavy loads analyses" and maintains the current safety analysis for the plant.

The NRC staff approved all TSUP Sections prior to the date of this letter, for Dresden Station, to be implemented by December 31, 1995. The current implementation schedule at Dresden Station, however, is dependent upon the startup from the current Dresden Unit 2 refueling outage, and a subsequent period of plant operation. The current startup schedule from the Unit 2 refueling outage is expected to be approximately January, 1996. To allow some margin for unforeseen changes in the startup and implementation schedule, therefore, ComEd requests a change to the implementation schedule for all approved TSUP Sections prior to the date of this letter, for Dresden Station, from December 31, 1995 until June 30, 1996. This proposed change is administrative in nature and does not adversely affect existing plant safety margins.

This supplemental application addresses all open items from all previously received NRC staff Open Items. A summary of the open items and ComEd's proposed resolutions are provided as an Attachment to this letter.

If there are any questions concerning this matter, please contact this office.

Sincerely,

Peter L. Piet
Nuclear Licensing Administrator

Attachment: A. Summary and Assessment of TSUP Clean-Up Changes
B. Marked-Up TSUP Pages
C. Revised TSUP Pages
D. Significant Hazards Evaluation of the Clean-Up Changes

cc: H. J. Miller, Regional Administrator - RIII
J. F. Stang, Project Manager - NRR
R. M. Pulsifer, Project Manager - NRR
C. L. Vanderniet, Senior Resident Inspector - Dresden
C. G. Miller, Senior Resident Inspector - Quad Cities
Office of Nuclear Facility Safety - IDNS

OFFICIAL SEAL
MARY JO YACK
NOTARY PUBLIC STATE OF ILLINOIS
MY COMMISSION EXPIRES 11/29/97

Mary Jo Yack 11-14-95

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REFERENCES

(1) P. Piet letter to T. Murley, dated July 29, 1992 (TSUP Sections 1.0, 3/4.0 and 3/4.3).

(2) P. Piet letter to T. Murley, dated September 15, 1992 (TSUP Sections 2.0, 3/4.11, and 3/4.12).

(3) P. Piet letter to T. Murley, dated October 15, 1992 (TSUP Sections 3/4.4).

(4) P. Piet letter to T. Murley, dated December 8, 1992 (TSUP Sections 3/4.1).

(5) P. Piet letter to T. Murley, dated January 14, 1993 (TSUP Implementation Information and Response to RAI on 3/4.0, 3/4.1, 3/4.3 and 3/4.10).

(6) P. Piet letter to T. Murley, dated February 16, 1993 (TSUP Sections 3/4.10).

(7) P. Piet letter to T. Murley, dated February 16, 1993 (TSUP Supplement to Sections 1.0, 3/4.0 and 3/4.3).

(8) P. Piet letter to T. Murley, dated March 9, 1993 (TSUP Supplement to Section 3/4.4).

(9) P. Piet letter to T. Murley, dated March 26, 1993 (TSUP Section 3/4.9).

(10) P. Piet letter to T. Murley, dated September 10, 1993 (TSUP Section 3/4.8).

(11) P. Piet letter to T. Murley, dated September 17, 1993 (TSUP Section 3/4.5).

(12) P. Piet letter to T. Murley, dated September 17, 1993 (TSUP Section 3/4.6).

(13) P. Piet letter to T. Murley, dated September 17, 1993 (TSUP Section 3/4.7).

(14) P. Piet letter to T. Murley, dated December 15, 1993 (TSUP Section 5.0).

(15) P. Piet letter to W. Russell, dated March 14, 1994 (TSUP Implementation Schedule).

(16) J.Schrage letter to W. Russell, dated August 30, 1994 (TSUP Section 3/4.2).

(17) P. Piet letter to U.S. NRC, dated January 27, 1995 (Changes to Section 3/4.0).

(18) J. Stang letter to D. Farrar, dated February 22, 1995 (TSUP NRC Staff RAI).

(19) J. Stang letter to D. Farrar, dated February 16, 1995 (NRC SER for TSUP Sections 1.0 and Section 3/4.0).

REFERENCES
(continued)

(20) P. Piet letter to U.S. NRC, dated April 21, 1995 (ComEd Response to NRC staff RAI for TSUP Section 5.0).

(21) P. Piet letter to U.S. NRC, dated April 21, 1995 (ComEd Response to NRC staff RAI for TSUP Section 2.0, 3/4.11 and 3/4.12).

(22) P. Piet letter to U.S. NRC, dated May 2, 1995 (ComEd Response to NRC staff RAI for TSUP Section 3/4.10).

(23) P. Piet letter to U.S. NRC, dated May 9, 1995 (ComEd Response to NRC staff RAI for TSUP Section 3/4.3).

(24) P. Piet letter to U.S. NRC, dated May 15, 1995 (ComEd Response to NRC staff RAI for TSUP Section 3/4.9).

(25) J. Schrage letter to U.S. NRC, dated May 17, 1995 (ComEd Response to NRC staff RAI for TSUP Section 3/4.1).

(26) J. Stang letter to D. Farrar, dated June 8, 1995 (NRC SER for TSUP Section 3/4.4).

(27) J. Stang letter to D. Farrar, dated June 13, 1995 (NRC SER for TSUP Sections 2.0, 3/4.11 and 3/4.12).

(28) J. Stang letter to D. Farrar, dated June 14, 1995 (NRC SER for TSUP Section 5.0).

(29) P. Piet letter to U.S. NRC, dated June 16, 1995 (ComEd Response to NRC staff RAI for TSUP Section 3/4.8).

(30) J. Stang letter to D. Farrar, dated June 23, 1995 (NRC SER for TSUP Section 3/4.10).

(31) P. Piet letter to U.S. NRC, dated June 30, 1995 (ComEd Response to NRC staff RAI for TSUP Section 3/4.6).

(32) P. Piet letter to U.S. NRC, dated July 20, 1995 (ComEd Response to NRC staff RAI for TSUP Section 3/4.7).

(33) J. Stang letter to D. Farrar, dated July 27, 1995 (NRC SER for TSUP Section 3/4.3).

(34) P. Piet letter to U.S. NRC, dated July 28, 1995 (ComEd Response to NRC staff RAI for TSUP Section 3/4.5).

(35) J. Schrage letter to U.S. NRC, dated August 4, 1995 (ComEd Response to NRC staff RAI for TSUP Section 3/4.2).

REFERENCES
(continued)

(36) J. Schrage letter to U.S. NRC, dated September 1, 1995 (Dresden TSUP Section 6.0).

(37) P. Piet letter to U.S. NRC, dated September 15, 1995 (TSUP Cleanup).

(38) J. Stang letter to D. Farrar, dated September 18, 1995 (NRC SER for TSUP Section 3/4.9).

(39) J. Stang letter to D. Farrar, dated September 20, 1995 (NRC SER for TSUP Section 3/4.1).

(40) J. Schrage letter to U.S. NRC, dated September 20, 1995 (Quad Cities TSUP Section 6.0).

(41) J. Stang letter to D. Farrar, dated September 21, 1995 (NRC SER for TSUP Section 3/4.6).

Commonwealth Edison Company
1400 Opus Place
Downers Grove, IL 60515

ComEd

November 14, 1995

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

SUBJECT: LaSalle County Nuclear Power Station Units 1 and 2
Dresden Nuclear Power Station Units 2 and 3
Quad Cities Nuclear Power Station Units 1 and 2
Request for Amendment to Facility Operating Licenses NPF-11, NPF-18, DPR-19, DPR-25, DPR-29 and DPR-30, Appendix A, Technical Specifications
Incorporation of Option B to 10CFR50, Appendix J
NRC Docket Nos. 50-373/374, 50-237/249 and 50-254/265

Pursuant to 10 CFR 50.90, ComEd proposes to amend Appendix A, Technical Specifications Section 3.6 of Facility Operating Licenses NPF-11 and NPF-18 and Technical Specifications Section 3.7 of Facility Operating Licenses DPR-19, DPR-25, DPR-29 and DPR-30. Due to the time frame of consideration, Dresden and Quad Cities' Technical Specification changes reflect the format of the Technical Specification Upgrade Program (TSUP).

The proposed Technical Specification Amendment is subdivided as follows:

1. Attachment A gives a description and safety analysis of the proposed changes.

2. Attachment B includes the proposed changes to the Technical Specifications pages, including marked-up versions of the current pages.

3. Attachment C describes ComEd's evaluation performed in accordance with 10 CFR 50.92 (c), which confirms that no significant hazards consideration is involved. In addition, ComEd's Environmental Assessment Applicability Review is included.

4. Attachment D describes the implementation plan for Option B to Appendix J for each site.

These proposed exemptions and Technical Specification amendment have been reviewed and approved by ComEd On-Site and Off-Site Review in accordance with ComEd procedures.

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ComEd requests NRC approval of this request prior to January 14, 1996 in order to adopt Option B to 10 CFR 50, Appendix J requirements at LaSalle County, Dresden, and Quad Cities Stations prior to upcoming refueling outages. In addition, the adoption of Option B to 10 CFR 50, Appendix J precludes the need for any on-going schedular exemptions required for Dresden Unit 3.

Based upon the costs incurred by current leak rate testing practices, ComEd believes the cost savings realized by the adoption of Option B to 10 CFR 50, Appendix J satisfies the criteria for Cost Beneficial Licensing Action (CBLA). Based upon the guidance provided in NUREG-1493, ComEd estimates cost savings well in excess of $100,000 individually for Dresden, LaSalle County and Quad Cities Stations by the incorporation of Option B to 10 CFR 50, Appendix J. Due to uncertainties associated with implementation of new requirements, exact cost savings are difficult to accurately predict at this time.

To the best of my knowledge and belief, the statements contained above are true and correct. In some respect these statements are not based on my personal knowledge, but obtained information furnished by other Commonwealth Edison employees, contractor employees, and consultants. Such information has been reviewed in accordance with company practice, and I believe it to be reliable.

ComEd is notifying the State of Illinois of this application for amendment by transmitting a copy of this letter and its attachments to the designated state official.

Please direct any questions you may have concerning this submittal to this office.

Sincerely,

Peter L. Piet

Peter L. Piet
Nuclear Licensing Administrator

Subscribed and Sworn to before me
on this 14th day of
November, 1995.
Mary Jo Yack
Notary Public

OFFICIAL SEAL
MARY JO YACK
NOTARY PUBLIC STATE OF ILLINOIS
MY COMMISSION EXPIRES 11/29/97

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Attachments:

A. Description and Safety Analysis of the Proposed Changes
B. Marked-Up Technical Specification Pages
C. Evaluation of Significant Hazards Considerations and Environmental Assessment Applicability Review
D. Site Implementation Plans for Option B to Appendix J

cc: H. J. Miller, Regional Administrator - RIII
P. G. Brochman, Senior Resident Inspector - LaSalle County
C. L. Vanderniet, Senior Resident Inspector - Dresden
C. G. Miller, Senior Resident Inspector - Quad Cities
R. Latta, Project Manager - NRR
J. Stang, Project Manager - NRR
R. Pulsifer, Project Manager - NRR
Office of Nuclear Facility Safety - IDNS

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Commonwealth Edison Company
Dresden Generating Station
6500 North Dresden Road
Morris, IL 60450
Tel 815-942-2920

ComEd

September 22, 1995

TPJLTR 95-0119

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Dresden Station Units 2 and 3
Clarification of Commitments Relating to Review of Dresden Station Technical Specifications.
Docket Nos. 50-237, 50-249

Reference:

a) T. P. Joyce letter to U. S. NRC dated June 2, 1995 transmitting Dresden Station's Response to SALP 13 Report

b) T. P. Joyce letter to U. S. NRC dated May 5, 1995 transmitting Dresden Station's Response to Notices of Violation issued in Inspection Report 50- 237/249/95004.

c) T. P. Joyce letter to U. S. NRC dated February 9, 1995 transmitting Dresden Station LER 237/95003.

The purpose of this letter is to provide clarification of the actions that are being be taken relative to our Technical Specification review identified in the referenced letters.

In references (b) and (c) we stated that "A team has been assembled to review the technical specifications with the intent of determining other inconsistencies similar to those existing in this event. In addition, the team will review training conducted on recent technical specification amendments to determine what additional operator training is necessary." In reference (b) we stated these actions would be completed by September 14, 1995.

In reference (a) we stated that "a comprehensive review of the Technical Specifications, their bases, operability evaluations, and operating procedures will be performed to ensure compliance of lower-tier documents with the Technical Specifications."

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Our plan is to conduct these reviews based on the new Technical Specifications, that will result from our Technical Specification Upgrade Project. This revision to our plan is based on the short time frame between completion of the review and implementation of the upgraded Technical Specifications.

This comprehensive review of the upgraded Technical Specifications is also serving as an independent readiness assessment of the new Technical Specifications. The review will provide assurance that the appropriate procedure changes have been accomplished, existing operability evaluations are in compliance with the new Technical Specifications, and the correct changes to USFAR have been identified.

The review of the upgraded Technical Specifications began in early August and will be completed by October 31, 1995. This is a change from the September 14, 1995 date stated in reference (b).

If there are any questions concerning this letter, please refer them to Peter Holland, Dresden Station Regulatory Assurance Supervisor, at (815) 942-2920, extension 2714.

Very truly yours,

FOR
Thomas P. Joyce
Site Vice President
Dresden Station

TPJ/kls

cc: H. J. Miller, Regional Administrator, Region III
W. T. Russell, Director, NRR
J. F. Stang, Project Manager, NRR (Unit 2/3)
C. L. Vanderniet, Senior Resident Inspector, Dresden Station
File: Numerical

Commonwealth Edison Company
1400 Opus Place
Downers Grove, IL 60515

ComEd

September 15, 1995

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Dresden Nuclear Power Station Units 2 and 3
Quad Cities Nuclear Power Station Units 1 and 2
Supplement to Application for Amendment to Facility Operating Licenses DPR-19, DPR-25, DPR-29 and DPR-30, Appendix A, Technical Specifications for the Technical Specifications Upgrade Program (TSUP)
NRC Docket Nos. 50-237/249 and 50-254/265

References: (see attached)

The purpose of this letter is to close out TSUP open items as identified by the NRC staff's review as noted in NRC staff Safety Evaluations received for previously provided submittals regarding the TSUP project (see attached References). A summary and ComEd's assessment of the proposed changes are provided as Attachment A to this letter. Attachment B highlights the proposed changes and includes marked-up versions of the affected TSUP pages. Attachment C provides revised TSUP pages reflecting the marked-up changes noted in Attachment B. Attachment D provides ComEd's supplemental evaluation of significant hazards considerations for the proposed resolution of the TSUP open items.

The proposed supplemental changes have been approved by Commonwealth Edison's (ComEd) Onsite and Offsite Review in accordance with Company procedures. Commonwealth Edison requests that the proposed changes be approved as submitted to become effective upon completion of the entire TSUP project.

To the best of my knowledge and belief, the statements contained in this document are true and correct. In some respects these statements are not based on my personal knowledge, but on information furnished by other ComEd employees, contractor employees, and/or consultants. Such information has been reviewed in accordance with company practice, and I believe it to be reliable.

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It should be noted that in Reference (a), the NRC staff approved TSUP Section 1.0 and Section 3/4.0 for both Dresden and Quad Cities to be implemented by December 31, 1995. The current implementation schedule at Quad Cities is February 1996. Therefore, ComEd requests change to the implementation schedule for Section 1.0 and Section 3/4.0 for Quad Cities to be changed from December 31, 1995 until June 30, 1996 to allow for any unforeseen changes in the schedule. This proposed change is administrative in nature. Further discussion of this change is provided as an attachment to this letter.

If there are any questions concerning this matter, please contact this office.

Sincerely,

Peter L. Piet
Nuclear Licensing Administrator

Attachment: A. Summary and Assessment of TSUP Clean-Up Changes
B. Marked-Up TSUP Pages
C. Revised TSUP Pages
D. Significant Hazards Evaluation of the Clean-Up Changes

cc: H. J. Miller, Regional Administrator - RIII
J. F. Stang, Project Manager - NRR
R. M. Pulsifer, Project Manager - NRR
C. L. Vanderniet, Senior Resident Inspector - Dresden
C. G. Miller, Senior Resident Inspector - Quad Cities
Office of Nuclear Facility Safety - IDNS

Signed before me on this 15th
day of September, 1995,
by Mary Jo Yack.
Notary Public

OFFICIAL SEAL
MARY JO YACK
NOTARY PUBLIC STATE OF ILLINOIS
MY COMMISSION EXPIRES 11/29/97

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REFERENCES

(a) J. Stang letter to D. Farrar, dated February 16, 1995 (NRC SER for TSUP Sections 1.0 and Section 3/4.0).

(b) J. Stang letter to D. Farrar, dated June 8, 1995 (NRC SER for TSUP Section 3/4.4).

(c) J. Stang letter to D. Farrar, dated June 14, 1995 (NRC SER for TSUP Section 5.0).

(d) J. Stang letter to D. Farrar, dated June 23, 1995 (NRC SER for TSUP Section 3/4.10).

(e) P. Piet letter to T. Murley, dated February 16, 1993 (TSUP Sections 3/4.10).

ATTACHMENT A

Summary and Assessment of TSUP Clean-Up Changes
Dresden and Quad Cities Nuclear Power Stations

No.	PAGES	TSUP SECTION	PLANT/DESCRIPTION
1	3/4.10-8, B 3/4.10-2	3/4.10.F Crane Travel	D/Q - changes to relocate TSUP 3/4.10.F (D/Q) and CTS 3.10.H for Dresden to administrative controls.
2	3/4.10-1, B 3/4.10-3	4.10.B Instrumentation	D/Q - eliminates 4.10.B, footnote (c) which allowed 0.7 cps with a signal-to-noise ratio of 2 for SRMs (vs. the 3 cps requirements).
3	1-5	RPS Response Times	D/Q - Add definition of RPS Response Time consistent with GL 93-08. Pages 1-6 and 1-7 included due to renumeration/shuffling of subsequent definitions.
4	5-1, 5-2, 5-3	Design Features	D/Q - Include text description of Exclusion Area and for the Low Population Zone.
5	License	DPR-29 & DPR-30	Quad Cities - Change implementation date from 12/31/95 until 6/30/96 for TSUP Sections 1.0 and 3/4.0.
6	NRC SER	3/4.4	D/Q - Clarify periodicity of TSUP 4.4.A.3.

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ATTACHMENT A

1. 3/4.10.F Crane Travel - Spent Fuel Storage Pool - In the NRC staff's Safety Evaluation Report (SER) for TSUP 3/4.10, "Refueling," dated June 23, 1995 (Reference (d)), Section 3.6 of the SER discussed TSUP 3/4.10.F, Crane Travel. The SER stated that a revised version of 3/4.10.F would be based on STS 3/4.9.7 and incorporate the loadings of the current TS (CTS 3.10.H for Dresden) requirements (loads no heavier than the weight of a single fuel assembly and handling tool). STS 4.9.7 provides the following guidelines: "Crane interlocks and physical stops which prevent crane travel with loads in excess of (1100) pounds over fuel assemblies in the spent fuel storage pool racks shall be demonstrated OPERABLE within 7 days prior to and at least once per 7 days during crane operation." The current Dresden and Quad Cities refueling crane/bridge design does not include such interlocks; as such, the prevention of transport of loads heavier than the weight of a single spent fuel assembly and handling tool are administratively controlled. Therefore, ComEd proposes that the requirements to control loads heavier than the weight of a single spent fuel assembly and handling tool be relocated to administrative controls. It should be noted that the proposed changes are consistent with the requirements of the Improved Standard Technical Specifications (NUREG-1433). Because administrative controls will continue to be enforced regarding the transport of loads heavier than the weight of a single spent fuel assembly and handling tool, existing plant safety margins are maintained. Therefore, ComEd considers this open item from the NRC staff's SER for TSUP 3/4.10 (Reference (d)) closed.

2. 4.10.B Instrumentation - This issue is applicable to both Dresden and Quad Cities and eliminates the proposed TSUP footnote (c) to SR 4.10.B which allowed 0.7 cps with a signal-to-noise (s/n) ratio of 2 for SRM operability. It should be noted that this issue was listed as an open item in the NRC staff's SER for TSUP 3/4.10 (Reference (d)). This change conservatively eliminates a less restrictive requirement from the proposed TSUP Section 4.10.B and is acceptable and consistent to the current licensing basis for Dresden and Quad Cities Stations. Therefore, ComEd considers this open item from the NRC staff's SER for TSUP 3/4.10 (Reference (d)) closed.

3. 1.0 REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIMES - To be consistent with the requirements of NUREG-0123, Dresden and Quad Cities are including the plant-specific definitions for REACTOR PROTECTION SYSTEM RESPONSE TIMES to Section 1.0 of TSUP. It should be noted that this issue was listed as an open item in the NRC staff's SER for TSUP 1.0 (Reference (a)). The existing definition for the Reactor Protection System response times from current Technical Specification 3.1.A.1 has been adopted in TSUP Section 1.0 as the Dresden and Quad Cities definition. To be consistent with the intention of Generic Letter 93-08, "Guidance for a Proposed License Amendment to Relocate Tables of Instrument Response Time Limits from Technical Specifications to the Updated Final Safety Analysis Report," any reference to specific instrumentation response shall be controlled within the UFSAR. As such, the proposed definition does not included specific time requirements. The change is equivalent to the current licensing basis for Dresden

and Quad Cities; therefore, there is no reduction in existing plant safety margins. Therefore, ComEd considers this open item from the NRC staff's SER for TSUP 1.0 (Reference (a)) closed.

4. 5.1 Design Features - Site - This issue is a proposed resolution to TSUP open item 3.9.1 and 3.9.2 of the NRC staff's SER for TSUP 5.0 (Reference (c)). The proposed clean-up package does not include figures for the Site Low Population Area and Exclusion Zone. ComEd proposed that information intended to be provided graphically in the figures are more properly controlled through the proposed TSUP textual description of this submittal. Any changes to the locations of the meteorological tower or effluent discharge points must conform to the requirements of 10 CFR 50.59. Furthermore, sufficient detail relating to these features exists in LCOs to ensure any changes which may affect safety require prior NRC review and approval. Features with a potential to affect safety are sufficiently addressed by LCOs. The proposed changes are administrative in nature as the proposed textual descriptions are the same as that found in the site UFSAR, therefore, the current licensing basis remains unchanged and the proposed clean-up changes are acceptable for TSUP 5.1. Therefore, ComEd considers the open items 3.9.1 and 3.9.2 from the NRC staff's SER for TSUP 5.0 (Reference (c)) closed.

5. Implementation Schedule - In Reference (a), the NRC staff approved TSUP Section 1.0 and Section 3/4.0 for both Dresden and Quad Cities to be implemented by December 31, 1995. The current implementation schedule at Quad Cities, however, is February, 1996. To allow some margin for unforeseen changes in the implementation schedule, therefore, ComEd requests a change to the implementation schedule for Section 1.0 and Section 3/4.0 for Quad Cities from December 31, 1995 until June 30, 1996. This proposed change is administrative in nature and does not adversely affect existing plant safety margins.

6. 3/4.4 NRC SER - In Reference (b), the NRC staff discussed the current test frequency for SBLC pumps (40 gpm per pump at 1275 psig) in TSUP 4.4.A.3 to be once every 31 days. As discussed in Attachment 5 of Reference (e), TSUP 4.4.A.3 replaced the current Technical Specification (CTS 4.4.A.1) monthly pump runs with quarterly (every 92 days) Inservice Testing (IST) provisions. These quarterly tests are in use at Dresden and Quad Cities and based upon experience, have adequately demonstrated system capabilities and availability. Therefore, TSUP 4.4.A.3 changes the frequency of the pump tests from every 31 days to every 92 days to be consistent with Dresden and Quad Cities IST program. Revisions to the IST program are controlled by the requirements of 10 CFR 50.55a. 10 CFR 50.55a provides sufficient controls to ensure the SBLC system pumps are adequately tested. Because the SBLC pumps are encompassed by the provisions of the IST program, existing plant safety margins are not significantly reduced by the proposed changes.

ATTACHMENT B

MARKED-UP TSUP PAGES
DRESDEN AND QUAD CITIES NUCLEAR POWER STATIONS
LICENSE NOS. DPR-19, DPR-25, DPR-29 AND DPR-30

Page	Applicable Plant
1-5	Dresden & Quad Cities
3/4.10-3	Dresden & Quad Cities
3/4.10-8	Dresden & Quad Cities
B 3/4.10-1	Dresden & Quad Cities
B 3/4.10-2	Dresden & Quad Cities
5-1	Dresden & Quad Cities
5-2	Dresden & Quad Cities
5-3	Dresden & Quad Cities

1.0 DEFINITIONS

PRIMARY CONTAINMENT INTEGRITY (PCI)

PRIMARY CONTAINMENT INTEGRITY (PCI) shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE primary containment automatic isolation valve system, or

2) Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except for valves that are open under administrative control as permitted by Specification 3.7.D.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.7.C.

d. The primary containment leakage rates are within the limits of Specification 3.7.B.

e. The suppression chamber is in compliance with the requirements of Specification 3.7.K.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM (PCP)

The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analysis, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

RATED THERMAL POWER (RTP)

RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of 2527 MWT.

REPORTABLE EVENT

A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME shall be the time interval for each trip function from the opening of the sensor contact up to and including the opening of the trip actuator.

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3.10 - LIMITING CONDITIONS FOR OPERATION

B. Instrumentation

At least 2 source range monitor[a] (SRM) CHANNEL(s) shall be OPERABLE and inserted to the normal operating level with:

1. Continuous visual indication in the control room,

2. One of the required SRM detectors located in the quadrant where CORE ALTERATION(s) are being performed and the other required SRM detector located in an adjacent quadrant, and

3. Unless adequate SHUTDOWN MARGIN has been demonstrated per Specification 3.3.A and the "one-rod-out" Refuel position interlock has been demonstrated OPERABLE per Specification 3.10.A, the "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn[b].

APPLICABILITY:

OPERATIONAL MODE 5, unless the following conditions are met:

1. No more than two fuel assemblies are present in each core quadrant associated with an SRM;

4.10 - SURVEILLANCE REQUIREMENTS

B. Instrumentation

Each of the required SRM channels shall be demonstrated OPERABLE by:

1. At least once per 12 hours:

   a. Performance of a CHANNEL CHECK.

   b. Verifying the detectors are inserted to the normal operating level, and

   c. During CORE ALTERATION(s), verifying that the detector of an OPERABLE SRM CHANNEL is located in the core quadrant where CORE ALTERATION(s) are being performed and another is located in an adjacent quadrant.

2. Performance of a CHANNEL FUNCTIONAL TEST:

   a. Within 24 hours prior to the start of CORE ALTERATION(s), and

   b. At least once per 7 days.

3. Verifying that the channel count rate is at least 3 cps~~(c)~~:

   a. Prior to control rod withdrawal,

   b. Prior to and at least once per 12 hours during CORE ALTERATION(s),

   c. At least once per 24 hours.

---

a The use of special movable detectors during CORE ALTERATION(s) in place of the normal SRM neutron detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

b Not required for control rods removed per Specification 3.10.I and 3.10.J

~~c May be reduced to 0.7 cps provided signal to noise ratio is greater than or equal to 2.0~~

3.10 - LIMITING CONDITIONS FOR OPERATION

F. ~~Crane Travel~~

All movements of a spent fuel shipping cask above the 545 foot elevation of the Reactor Building shall be controlled by the "Restricted Mode" path control system of the reactor building crane.

APPLICABILITY:

At all times.

ACTION:

With the requirements of the above specification not satisfied:

1. Operation may continue with a failed controlled area limit switch for 48 hours provided an operator is on the refueling floor to assure the reactor building crane is operated within the restricted zone painted on the floor, or
2. Place the crane load in a safe condition.

The provisions of Specification 3.0.C are not applicable.

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4.10 - SURVEILLANCE REQUIREMENTS

F. ~~Crane Travel~~

1. The spent fuel shipping cask "Restricted Mode" path control system of the reactor building crane shall be demonstrated OPERABLE within 7 days prior to and at least once per 7 days during spent fuel shipping cask movement over the refueling floor.
2. The redundant crane including the rope, hooks, slings, shackles and other operating mechanisms shall be inspected prior to spent fuel shipping cask handling operations and the rope will be replaced if any of the following conditions exist:
   a. Twelve randomly distributed broken wires in one lay or four broken wires in one strand of one rope lay.
   b. Wear of one-third of the original diameter of outside individual wire.
   c. Kinking, crushing, or any other damage resulting in distortion of the rope.
   d. Evidence of any type of heat damage.
   e. Reductions from nominal diameter of more than 1/16 inch for a rope diameter from 7/8 inch to 1-1/4 inch inclusive.
3. The spent fuel cask will be lifted free of all support by a maximum of 1 foot and left hanging for 5 minutes prior to spent fuel cask handling operations.

BASES

3/4.10.A Reactor Mode Switch

Locking the OPERABLE reactor mode switch in the Shutdown or Refuel position, as specified, ensures that the restrictions on control rod withdrawal and refueling platform movement during the refueling operations are properly activated. These conditions reinforce the refueling procedures and reduce the probability of inadvertent criticality, damage to reactor internals or fuel assemblies, and exposure of personnel to excessive radioactivity.

The addition of large amounts of reactivity to the core is prevented by operating procedures, which are in turn backed up by refueling interlocks on rod withdrawal and movement of the refueling platform. When the mode switch is in the Refuel position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist. If the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. With the mode switch in the refuel position only one control rod can be withdrawn.

3/4.10.B Instrumentation

The OPERABILITY of at least two source range monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core, whenever reactor criticality is possible.

The source range monitors (SRM) are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and reactor startup. Requiring two OPERABLE source range monitors in and adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations. Requiring a minimum of 3 counts per second whenever criticality is possible provides assurance that neutron flux is being monitored. The SRM system is designed to provide a signal-to-noise ratio of at least 3:1 and a count rate of at least 3 counts per second. ~~For a signal-to-noise ratio of 2:1, the count rate must be at least 0.7 counts per second.~~ Criticality is considered to be impossible if there are no more than two assemblies in a quadrant and if these are in locations adjacent to the source range monitors (i.e., spatially separated).

Special movable detectors may be used during CORE ALTERATION(s) in place of the normal SRM neutron detectors. These special detectors must be connected to the normal SRM circuits such that the applicable neutron flux indication, control rod blocks and scram signals can be generated. The special detectors provide more flexibility in monitoring reactivity changes during fuel loading since they can be positioned anywhere within the core during refueling provided they meet the location requirements of the specification.

When the Reactor Protection System shorting links are removed, the source range monitors provide added protection against local criticalities by providing an initiating signal for a reactor scram on high neutron flux.

BASES

3/4.10.C Control Rod Position

The requirement that all control rods be inserted during other CORE ALTERATION(s) ensures that fuel will not be loaded into a cell without an inserted control rod.

3/4.10.D Decay Time

The minimum requirement for reactor subcriticality prior to fuel movement ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.10.E Communications

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status regarding core reactivity conditions during movement of fuel within the reactor pressure vessel.

3/4.10.F ~~Crane Travel~~ DELETED

~~The operation of the reactor building crane in the Restricted Mode during spent fuel shipping cask handling operations, assures that the cask remains within the controlled area once it has been removed from its transport vehicle. The surveillance requirements specified assure that the crane is adequately inspected in accordance with the accepted ANSI Standard (B.30.2.0) and the manufacturer's recommendations to determine that the equipment is in satisfactory condition. The testing of the controlled area limit switches assures that the crane operation will be limited to the designated area in the Restricted Mode of operation. Requiring the lifting and holding of the cask for 5 minutes during the initial lift of cask handling operations puts a load test on the entire crane lifting mechanism as well as the braking system. Performing this test when the cask is being lifted initially assures that the system is OPERABLE prior to lifting the load to excessive height.~~

3/4.10.G Water Level - Reactor Vessel

3/4.10.H Water Level - Spent Fuel Storage Pool

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. This minimum water depth is consistent with the assumptions of the accident analysis.

5.0 DESIGN FEATURES

5.1 SITE

Site and Exclusion Area

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5.1.A [INTENTIONALLY BLANK]

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Low Population Zone

5.1.B The low population zone shall be as shown in Figure 5.1.B-1.

a five mile radius from the centerline of the chimney.

Radioactive Gaseous Effluents

5.1.C Information regarding radioactive gaseous effluents shall be located in the OFFSITE DOSE CALCULATION MANUAL.

Radioactive Liquid Effluents

5.1.D Information regarding radioactive liquid effluents shall be located in the OFFSITE DOSE CALCULATION MANUAL.

INSERT

The site consists of approximately 953 acres adjacent to the Illinois River at the point where it is formed by the confluence of the Des Plaines and Kankakee Rivers, in the northeast quarter of the Goose Lake Township, Grundy County, Illinois. The Exclusion Area shall not be less than 800 meters from the centerline of the reactor vessels.

FIGURE 5.1.A-1

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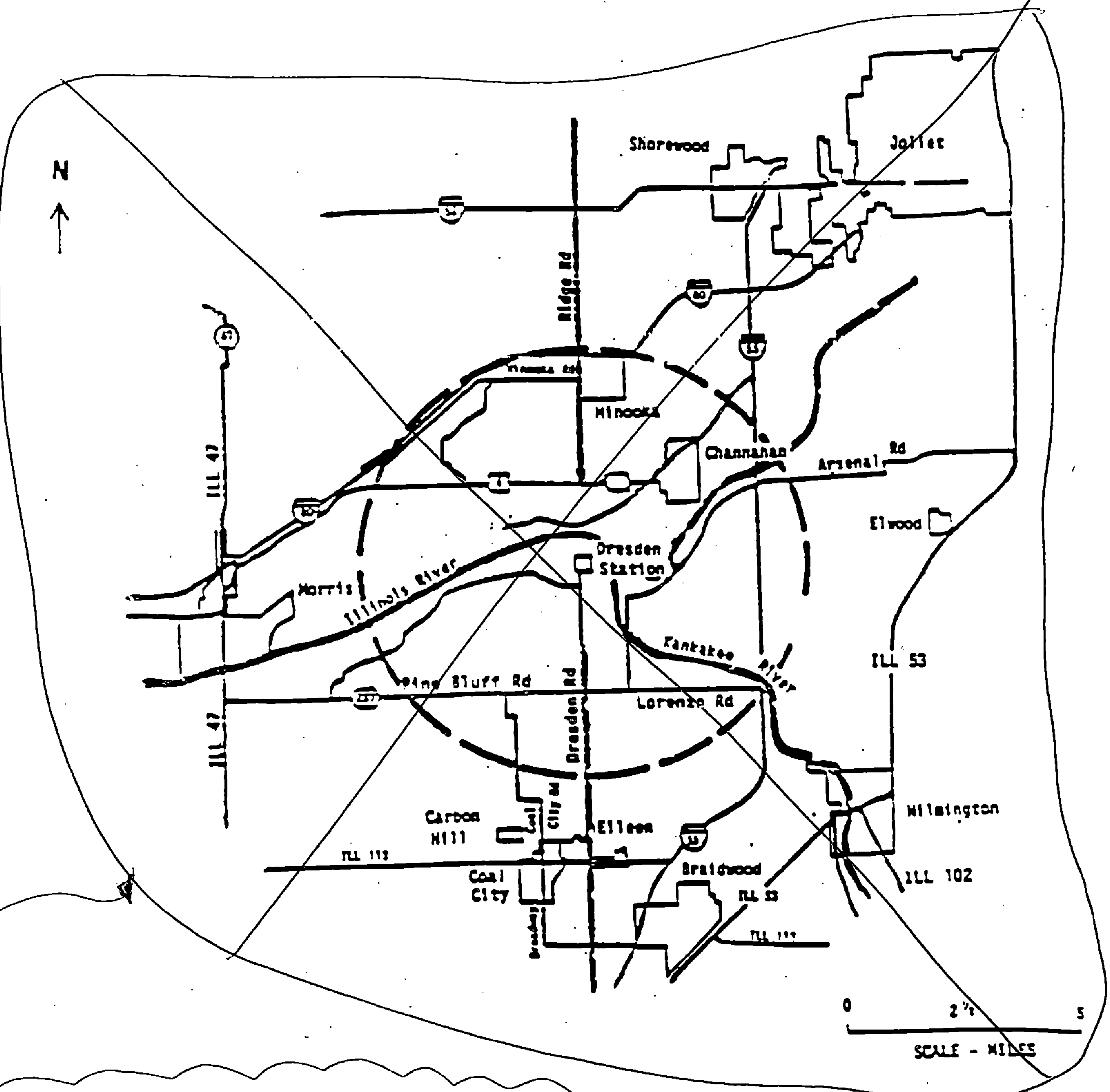
FIGURE 5.1.B-1

LOW POPULATION ZONE



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1.0 DEFINITIONS

PRIMARY CONTAINMENT INTEGRITY (PCI)

PRIMARY CONTAINMENT INTEGRITY (PCI) shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE primary containment automatic isolation valve system, or

2) Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except for valves that are open under administrative control as permitted by Specification 3.7.D.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.7.C.

d. The primary containment leakage rates are within the limits of Specification 3.7.B.

e. The suppression chamber is in compliance with the requirements of Specification 3.7.K.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM (PCP)

The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analysis, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

RATED THERMAL POWER (RTP)

RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of 2511 MWT.

REPORTABLE EVENT

A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME shall be the time interval for each trip function from the opening of the sensor contact up to and including the opening of the trip actuator.

3.10 - LIMITING CONDITIONS FOR OPERATION

B. Instrumentation

At least 2 source range monitor[a] (SRM) CHANNEL(s) shall be OPERABLE and inserted to the normal operating level with:

1. Continuous visual indication in the control room,

2. One of the required SRM detectors located in the quadrant where CORE ALTERATION(s) are being performed and the other required SRM detector located in an adjacent quadrant, and

3. Unless adequate SHUTDOWN MARGIN has been demonstrated per Specification 3.3.A and the "one-rod-out" Refuel position interlock has been demonstrated OPERABLE per Specification 3.10.A, the "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn[b].

APPLICABILITY:

OPERATIONAL MODE 5, unless the following conditions are met:

1. No more than two fuel assemblies are present in each core quadrant associated with an SRM;

4.10 - SURVEILLANCE REQUIREMENTS

B. Instrumentation

Each of the required SRM channels shall be demonstrated OPERABLE by:

1. At least once per 12 hours:

   a. Performance of a CHANNEL CHECK.

   b. Verifying the detectors are inserted to the normal operating level, and

   c. During CORE ALTERATION(s), verifying that the detector of an OPERABLE SRM CHANNEL is located in the core quadrant where CORE ALTERATION(s) are being performed and another is located in an adjacent quadrant.

2. Performance of a CHANNEL FUNCTIONAL TEST:

   a. Within 24 hours prior to the start of CORE ALTERATION(s), and

   b. At least once per 7 days.

3. Verifying that the channel count rate is at least 3 cps~~[c]:~~

   a. Prior to control rod withdrawal,

   b. Prior to and at least once per 12 hours during CORE ALTERATION(s),

   c. At least once per 24 hours.

---

a The use of special movable detectors during CORE ALTERATION(s) in place of the normal SRM neutron detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

b Not required for control rods removed per Specification 3.10.I and 3.10.J

~~c May be reduced to 0.7 cps provided signal to noise ratio is greater than or equal to 2.0.~~

3.10 - LIMITING CONDITIONS FOR OPERATION

F. ~~Crane Travel~~

All movements of a spent fuel shipping cask above the 623 foot elevation of the Reactor Building shall be controlled by the "Restricted Mode" path control system of the reactor building crane.

APPLICABILITY:

At all times.

ACTION:

With the requirements of the above specification not satisfied:

1. Operation may continue with a failed controlled area limit switch for 48 hours provided an operator is on the refueling floor to assure the reactor building crane is operated within the restricted zone painted on the floor, or
2. Place the crane load in a safe condition.

The provisions of Specification 3.0.C are not applicable.

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4.10 - SURVEILLANCE REQUIREMENTS

F. ~~Crane Travel~~

1. The spent fuel shipping cask "Restricted Mode" path control system of the reactor building crane shall be demonstrated OPERABLE within 7 days prior to and at least once per 7 days during spent fuel shipping cask movement above the 623 foot elevation of the reactor building.
2. The redundant crane including the rope, hooks, slings, shackles and other operating mechanisms shall be inspected prior to spent fuel shipping cask handling operations and the rope will be replaced if any of the following conditions exist:
   a. Twelve randomly distributed broken wires in one lay or four broken wires in one strand of one rope lay.
   b. Wear of one-third of the original diameter of outside individual wire.
   c. Kinking, crushing, or any other damage resulting in distortion of the rope.
   d. Evidence of any type of heat damage.
   e. Reductions from nominal diameter of more than 1/16 inch for a rope diameter from 7/8 inch to 1-1/4 inch inclusive.
3. The spent fuel cask will be lifted free of all support by a maximum of 1 foot and left hanging for 5 minutes prior to spent fuel cask handling operations.

BASES

3/4.10.A Reactor Mode Switch

Locking the OPERABLE reactor mode switch in the Shutdown or Refuel position, as specified, ensures that the restrictions on control rod withdrawal and refueling platform movement during the refueling operations are properly activated. These conditions reinforce the refueling procedures and reduce the probability of inadvertent criticality, damage to reactor internals or fuel assemblies, and exposure of personnel to excessive radioactivity.

The addition of large amounts of reactivity to the core is prevented by operating procedures, which are in turn backed up by refueling interlocks on rod withdrawal and movement of the refueling platform. When the mode switch is in the Refuel position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist. If the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. With the mode switch in the refuel position only one control rod can be withdrawn.

3/4.10.B Instrumentation

The OPERABILITY of at least two source range monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core, whenever reactor criticality is possible.

The source range monitors (SRM) are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and reactor startup. Requiring two OPERABLE source range monitors in and adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations. Requiring a minimum of 3 counts per second whenever criticality is possible provides assurance that neutron flux is being monitored. The SRM system is designed to provide a signal-to-noise ratio of at least 3:1 and a count rate of at least 3 counts per second. ~~For a signal-to-noise ratio of 2:1, the count rate must be at least 0.7 counts per second.~~ Criticality is considered to be impossible if there are no more than two assemblies in a quadrant and if these are in locations adjacent to the source range monitors (i.e., spatially separated).

Special movable detectors may be used during CORE ALTERATION(s) in place of the normal SRM neutron detectors. These special detectors must be connected to the normal SRM circuits such that the applicable neutron flux indication, control rod blocks and scram signals can be generated. The special detectors provide more flexibility in monitoring reactivity changes during fuel loading since they can be positioned anywhere within the core during refueling provided they meet the location requirements of the specification.

When the Reactor Protection System shorting links are removed, the source range monitors provide added protection against local criticalities by providing an initiating signal for a reactor scram on high neutron flux.

BASES

3/4.10.C Control Rod Position

The requirement that all control rods be inserted during other CORE ALTERATION(s) ensures that fuel will not be loaded into a cell without an inserted control rod.

3/4.10.D Decay Time

The minimum requirement for reactor subcriticality prior to fuel movement ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.10.E Communications

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status regarding core reactivity conditions during movement of fuel within the reactor pressure vessel.

3/4.10.F ~~Crane Travel~~ DELETED

~~The operation of the reactor building crane in the Restricted Mode during spent fuel shipping cask handling operations assures that the cask remains within the controlled area once it has been removed from its transport vehicle. The surveillance requirements specified assure that the crane is adequately inspected in accordance with the accepted ANSI Standard (B.30.2.0) and the manufacturer's recommendations to determine that the equipment is in satisfactory condition. The testing of the controlled area limit switches assures that the crane operation will be limited to the designated area in the Restricted Mode of operation. Requiring the lifting and holding of the cask for 5 minutes during the initial lift of cask handling operations puts a load test on the entire crane lifting mechanism as well as the braking system. Performing this test when the cask is being lifted initially assures that the system is OPERABLE prior to lifting the load to excessive height.~~

3/4.10.G Water Level - Reactor Vessel

3/4.10.H Water Level - Spent Fuel Storage Pool

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. This minimum water depth is consistent with the assumptions of the accident analysis.

5.0 DESIGN FEATURES

5.1 SITE

Site and Exclusion Area

~~[INTENTIONALLY BLANK]~~

5.1.A ~~[INTENTIONALLY BLANK]~~ "INSERT 7"

Low Population Zone

5.1.B The low population zone shall be ~~as shown in Figure 5.1.B-1~~ a three mile radius from the centerline of the chimney.

Radioactive Gaseous Effluents

5.1.C Information regarding radioactive gaseous effluents shall be located in the OFFSITE DOSE CALCULATION MANUAL.

Radioactive Liquid Effluents

5.1.D Information regarding radioactive liquid effluents shall be located in the OFFSITE DOSE CALCULATION MANUAL.

INSERT

The site consists of approximately 784 acres on the east bank of the Mississippi River opposite the mouth of the Wapsipinicon River, approximately three miles north of the village of Cordova, Rock Island County, Illinois. The Exclusion Area shall not be less than 380 meters from the centerline of the chimney.

FIGURE 5.1.A-1

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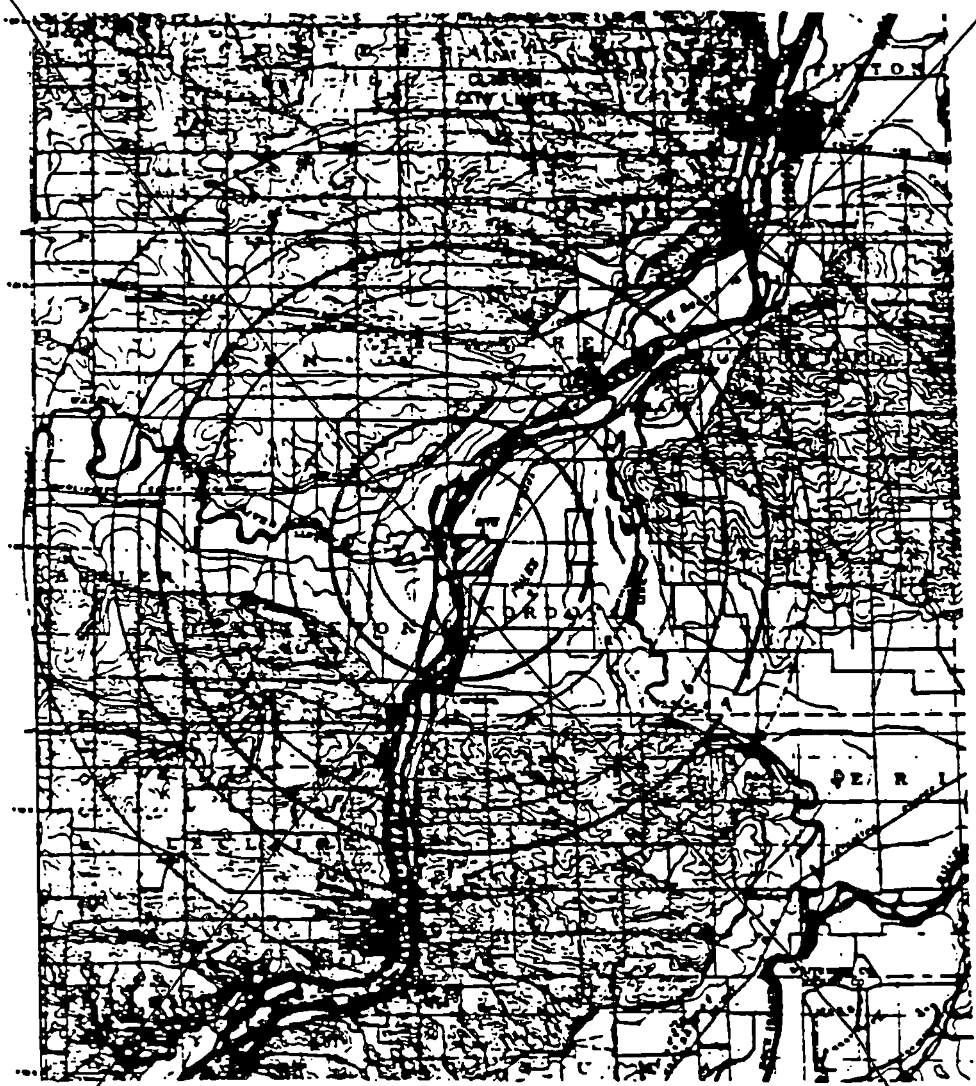
FIGURE 5.1.B-1

LOW POPULATION ZONE



POPULATION CENTER DISTANCE, 7 MILE RADIUS
LOW POPULATION ZONE, 3 MILE RADIUS

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ATTACHMENT C

REVISED TSUP PAGES FOR
DRESDEN AND QUAD CITIES NUCLEAR POWER STATIONS
LICENSE NOS. DPR-19, DPR-25, DPR-29, AND DPR-30

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1.0 DEFINITIONS

PRIMARY CONTAINMENT INTEGRITY (PCI)

PRIMARY CONTAINMENT INTEGRITY (PCI) shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

   1) Capable of being closed by an OPERABLE primary containment automatic isolation valve system, or

   2) Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except for valves that are open under administrative control as permitted by Specification 3.7.D.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.7.C.

d. The primary containment leakage rates are within the limits of Specification 3.7.B.

e. The suppression chamber is in compliance with the requirements of Specification 3.7.K.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM (PCP)

The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analysis, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

RATED THERMAL POWER (RTP)

RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of 2527 MWT.

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME shall be the time interval for each trip function from the opening of the sensor contact up to and including the opening of the trip actuator.

1.0 DEFINITIONS

REPORTABLE EVENT

A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SECONDARY CONTAINMENT INTEGRITY (SCI)

SECONDARY CONTAINMENT INTEGRITY (SCI) shall exist when:

a. All secondary containment penetrations required to be closed during accident conditions are either:

   1) Capable of being closed by an OPERABLE secondary containment automatic isolation valve system, or

   2) Closed by at least one manual valve, blind flange, or deactivated automatic damper secured in its closed position, except as permitted by Specification 3.7.O.

b. All secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.7.P.

d. At least one door in each access to the secondary containment is closed.

e. The sealing mechanism associated with each secondary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

f. The pressure within the secondary containment is less than or equal to the value required by Specification 4.7.N.1.

SHUTDOWN MARGIN (SDM)

SHUTDOWN MARGIN (SDM) shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F; and xenon free.

SOURCE CHECK

A SOURCE CHECK shall be the qualitative assessment of CHANNEL response when the CHANNEL sensor is exposed to a radioactive source.

STEADY STATE LINEAR HEAT GENERATION RATE (SLHGR)

The STEADY STATE LINEAR HEAT GENERATION RATE (SLHGR) shall be the limit which protects against exceeding the fuel end-of-life steady state design criteria.

1.0 DEFINITIONS

THERMAL POWER

THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TRANSIENT LINEAR HEAT GENERATION RATE (TLHGR)

The TRANSIENT LINEAR HEAT GENERATION RATE (TLHGR) shall be the limit which protects against fuel centerline melting and 1% plastic cladding strain during transient conditions throughout the life of the fuel.

TRIP SYSTEM

A TRIP SYSTEM shall be an arrangement of instrument CHANNEL trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A TRIP SYSTEM may require one or more instrument CHANNEL trip signals related to one or more plant parameters in order to initiate TRIP SYSTEM action. Initiation of protective action may require the tripping of a single TRIP SYSTEM or the coincident tripping of two TRIP SYSTEMs.

UNIDENTIFIED LEAKAGE

UNIDENTIFIED LEAKAGE shall be all leakage in the primary containment which is not IDENTIFIED LEAKAGE.

3.10 - LIMITING CONDITIONS FOR OPERATION

B. Instrumentation

At least 2 source range monitor[a] (SRM) CHANNEL(s) shall be OPERABLE and inserted to the normal operating level with:

1. Continuous visual indication in the control room,

2. One of the required SRM detectors located in the quadrant where CORE ALTERATION(s) are being performed and the other required SRM detector located in an adjacent quadrant, and

3. Unless adequate SHUTDOWN MARGIN has been demonstrated per Specification 3.3.A and the "one-rod-out" Refuel position interlock has been demonstrated OPERABLE per Specification 3.10.A, the "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn[b].

APPLICABILITY:

OPERATIONAL MODE 5, unless the following conditions are met:

1. No more than two fuel assemblies are present in each core quadrant associated with an SRM;

4.10 - SURVEILLANCE REQUIREMENTS

B. Instrumentation

Each of the required SRM channels shall be demonstrated OPERABLE by:

1. At least once per 12 hours:

   a. Performance of a CHANNEL CHECK.

   b. Verifying the detectors are inserted to the normal operating level, and

   c. During CORE ALTERATION(s), verifying that the detector of an OPERABLE SRM CHANNEL is located in the core quadrant where CORE ALTERATION(s) are being performed and another is located in an adjacent quadrant.

2. Performance of a CHANNEL FUNCTIONAL TEST:

   a. Within 24 hours prior to the start of CORE ALTERATION(s), and

   b. At least once per 7 days.

3. Verifying that the channel count rate is at least 3 cps:

   a. Prior to control rod withdrawal,

   b. Prior to and at least once per 12 hours during CORE ALTERATION(s),

   c. At least once per 24 hours.

---

a The use of special movable detectors during CORE ALTERATION(s) in place of the normal SRM neutron detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

b Not required for control rods removed per Specification 3.10.I and 3.10.J

3.10 - LIMITING CONDITIONS FOR OPERATION

F. DELETED

4.10 - SURVEILLANCE REQUIREMENTS

F. DELETED

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BASES

3/4.10.A Reactor Mode Switch

Locking the OPERABLE reactor mode switch in the Shutdown or Refuel position, as specified, ensures that the restrictions on control rod withdrawal and refueling platform movement during the refueling operations are properly activated. These conditions reinforce the refueling procedures and reduce the probability of inadvertent criticality, damage to reactor internals or fuel assemblies, and exposure of personnel to excessive radioactivity.

The addition of large amounts of reactivity to the core is prevented by operating procedures, which are in turn backed up by refueling interlocks on rod withdrawal and movement of the refueling platform. When the mode switch is in the Refuel position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist. If the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. With the mode switch in the refuel position only one control rod can be withdrawn.

3/4.10.B Instrumentation

The OPERABILITY of at least two source range monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core, whenever reactor criticality is possible.

The source range monitors (SRM) are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and reactor startup. Requiring two OPERABLE source range monitors in and adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations. Requiring a minimum of 3 counts per second whenever criticality is possible provides assurance that neutron flux is being monitored. The SRM system is designed to provide a signal-to-noise ratio of at least 3:1 and a count rate of at least 3 counts per second. Criticality is considered to be impossible if there are no more than two assemblies in a quadrant and if these are in locations adjacent to the source range monitors (i.e., spatially separated).

Special movable detectors may be used during CORE ALTERATION(s) in place of the normal SRM neutron detectors. These special detectors must be connected to the normal SRM circuits such that the applicable neutron flux indication, control rod blocks and scram signals can be generated. The special detectors provide more flexibility in monitoring reactivity changes during fuel loading since they can be positioned anywhere within the core during refueling provided they meet the location requirements of the specification.

When the Reactor Protection System shorting links are removed, the source range monitors provide added protection against local criticalities by providing an initiating signal for a reactor scram on high neutron flux.

BASES

3/4.10.C Control Rod Position

The requirement that all control rods be inserted during other CORE ALTERATION(s) ensures that fuel will not be loaded into a cell without an inserted control rod.

3/4.10.D Decay Time

The minimum requirement for reactor subcriticality prior to fuel movement ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.10.E Communications

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status regarding core reactivity conditions during movement of fuel within the reactor pressure vessel.

3/4.10.F DELETED

3/4.10.G Water Level - Reactor Vessel

3/4.10.H Water Level - Spent Fuel Storage Pool

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. This minimum water depth is consistent with the assumptions of the accident analysis.

5.0 DESIGN FEATURES

5.1 SITE

Site and Exclusion Area

5.1.A The site consists of approximately 953 acres adjacent to the Illinois River at the point where it is formed by the confluence of the Des Plaines and Kankakee Rivers, in the northeast quarter of the Goose Lake Township, Grundy County, Illinois. The Exclusion Area shall not be less than 800 meters from the centerline of the chimney.

Low Population Zone

5.1.B The Low Population Zone shall be a five mile radius from the centerline of the chimney.

Radioactive Gaseous Effluents

5.1.C Information regarding radioactive gaseous effluents shall be located in the OFFSITE DOSE CALCULATION MANUAL.

Radioactive Liquid Effluents

5.1.D Information regarding radioactive liquid effluents shall be located in the OFFSITE DOSE CALCULATION MANUAL.

FIGURE 5.1.A-1

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FIGURE 5.1.B-1

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1.0 DEFINITIONS

PRIMARY CONTAINMENT INTEGRITY (PCI)

PRIMARY CONTAINMENT INTEGRITY (PCI) shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

   1) Capable of being closed by an OPERABLE primary containment automatic isolation valve system, or

   2) Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except for valves that are open under administrative control as permitted by Specification 3.7.D.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.7.C.

d. The primary containment leakage rates are within the limits of Specification 3.7.B.

e. The suppression chamber is in compliance with the requirements of Specification 3.7.K.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM (PCP)

The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analysis, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

RATED THERMAL POWER (RTP)

RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of 2511 MWT.

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME shall be the time interval for each trip function from the opening of the sensor contact up to and including the opening of the trip actuator.

1.0 DEFINITIONS

REPORTABLE EVENT

A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY (SCI)

SECONDARY CONTAINMENT INTEGRITY (SCI) shall exist when:

a. All secondary containment penetrations required to be closed during accident conditions are either:

   1) Capable of being closed by an OPERABLE secondary containment automatic isolation valve system, or

   2) Closed by at least one manual valve, blind flange, or deactivated automatic damper secured in its closed position, except as permitted by Specification 3.7.O.

b. All secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.7.P.

d. At least one door in each access to the secondary containment is closed.

e. The sealing mechanism associated with each secondary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

f. The pressure within the secondary containment is less than or equal to the value required by Specification 4.7.N.1.

SHUTDOWN MARGIN (SDM)

SHUTDOWN MARGIN (SDM) shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F; and xenon free.

SOURCE CHECK

A SOURCE CHECK shall be the qualitative assessment of CHANNEL response when the CHANNEL sensor is exposed to a radioactive source.

1.0 DEFINITIONS

THERMAL POWER
THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TRIP SYSTEM
A TRIP SYSTEM shall be an arrangement of instrument CHANNEL trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A TRIP SYSTEM may require one or more instrument CHANNEL trip signals related to one or more plant parameters in order to initiate TRIP SYSTEM action. Initiation of protective action may require the tripping of a single TRIP SYSTEM or the coincident tripping of two TRIP SYSTEMs.

UNIDENTIFIED LEAKAGE
UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

3.10 - LIMITING CONDITIONS FOR OPERATION

B. Instrumentation

At least 2 source range monitor[a] (SRM) CHANNEL(s) shall be OPERABLE and inserted to the normal operating level with:

1. Continuous visual indication in the control room,

2. One of the required SRM detectors located in the quadrant where CORE ALTERATION(s) are being performed and the other required SRM detector located in an adjacent quadrant, and

3. Unless adequate SHUTDOWN MARGIN has been demonstrated per Specification 3.3.A and the "one-rod-out" Refuel position interlock has been demonstrated OPERABLE per Specification 3.10.A, the "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn[b].

APPLICABILITY:

OPERATIONAL MODE 5, unless the following conditions are met:

1. No more than two fuel assemblies are present in each core quadrant associated with an SRM;

4.10 - SURVEILLANCE REQUIREMENTS

B. Instrumentation

Each of the required SRM channels shall be demonstrated OPERABLE by:

1. At least once per 12 hours:

   a. Performance of a CHANNEL CHECK.

   b. Verifying the detectors are inserted to the normal operating level, and

   c. During CORE ALTERATION(s), verifying that the detector of an OPERABLE SRM CHANNEL is located in the core quadrant where CORE ALTERATION(s) are being performed and another is located in an adjacent quadrant.

2. Performance of a CHANNEL FUNCTIONAL TEST:

   a. Within 24 hours prior to the start of CORE ALTERATION(s), and

   b. At least once per 7 days.

3. Verifying that the channel count rate is at least 3 cps:

   a. Prior to control rod withdrawal,

   b. Prior to and at least once per 12 hours during CORE ALTERATION(s),

   c. At least once per 24 hours.

---

a The use of special movable detectors during CORE ALTERATION(s) in place of the normal SRM neutron detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

b Not required for control rods removed per Specification 3.10.I and 3.10.J

3.10 - LIMITING CONDITIONS FOR OPERATION	4.10 - SURVEILLANCE REQUIREMENTS
F. DELETED	F. DELETED

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BASES

3/4.10.A Reactor Mode Switch

Locking the OPERABLE reactor mode switch in the Shutdown or Refuel position, as specified, ensures that the restrictions on control rod withdrawal and refueling platform movement during the refueling operations are properly activated. These conditions reinforce the refueling procedures and reduce the probability of inadvertent criticality, damage to reactor internals or fuel assemblies, and exposure of personnel to excessive radioactivity.

The addition of large amounts of reactivity to the core is prevented by operating procedures, which are in turn backed up by refueling interlocks on rod withdrawal and movement of the refueling platform. When the mode switch is in the Refuel position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist. If the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. With the mode switch in the refuel position only one control rod can be withdrawn.

3/4.10.B Instrumentation

The OPERABILITY of at least two source range monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core, whenever reactor criticality is possible.

The source range monitors (SRM) are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and reactor startup. Requiring two OPERABLE source range monitors in and adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations. Requiring a minimum of 3 counts per second whenever criticality is possible provides assurance that neutron flux is being monitored. The SRM system is designed to provide a signal-to-noise ratio of at least 3:1 and a count rate of at least 3 counts per second. Criticality is considered to be impossible if there are no more than two assemblies in a quadrant and if these are in locations adjacent to the source range monitors (i.e., spatially separated).

Special movable detectors may be used during CORE ALTERATION(s) in place of the normal SRM neutron detectors. These special detectors must be connected to the normal SRM circuits such that the applicable neutron flux indication, control rod blocks and scram signals can be generated. The special detectors provide more flexibility in monitoring reactivity changes during fuel loading since they can be positioned anywhere within the core during refueling provided they meet the location requirements of the specification.

When the Reactor Protection System shorting links are removed, the source range monitors provide added protection against local criticalities by providing an initiating signal for a reactor scram on high neutron flux.

BASES

3/4.10.C Control Rod Position

The requirement that all control rods be inserted during other CORE ALTERATION(s) ensures that fuel will not be loaded into a cell without an inserted control rod.

3/4.10.D Decay Time

The minimum requirement for reactor subcriticality prior to fuel movement ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.10.E Communications

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status regarding core reactivity conditions during movement of fuel within the reactor pressure vessel.

3/4.10.F DELETED

3/4.10.G Water Level - Reactor Vessel

3/4.10.H Water Level - Spent Fuel Storage Pool

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. This minimum water depth is consistent with the assumptions of the accident analysis.

5.0 DESIGN FEATURES

5.1 SITE

Site and Exclusion Area

5.1.A The site consists of approximately 784 acres on the east bank of the Mississippi River opposite the mouth of the Wapsipinicon River, approximately three miles north of the village of Cordova, Rock Island County, Illinois. The Exclusion Area shall not be less than 380 meters from the centerline of the chimney.

Low Population Zone

5.1.B The Low Population Zone shall be a three mile radius from the centerline of the chimney.

Radioactive Gaseous Effluents

5.1.C Information regarding radioactive gaseous effluents shall be located in the OFFSITE DOSE CALCULATION MANUAL.

Radioactive Liquid Effluents

5.1.D Information regarding radioactive liquid effluents shall be located in the OFFSITE DOSE CALCULATION MANUAL.

FIGURE 5.1.A-1

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ATTACHMENT D

SIGNIFICANT HAZARDS EVALUATION FOR DRESDEN AND QUAD CITIES NUCLEAR POWER STATIONS FOR LICENSE NOS. DPR-19, DPR-25, DPR-29, AND DPR-30

ComEd has evaluated this proposed amendment **that resolves open items from the Technical Specification Upgrade Program (TSUP)** and determined that it involves no significant hazards consideration. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility, in accordance with the proposed amendment, would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3) Involve a significant reduction in a margin of safety.

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because:

In general, the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Implementation of these changes will provide increased reliability of equipment assumed to operate in the current safety analysis, or provide continued assurance that specified parameters remain within their acceptance limits, and as such, will not significantly increase the probability or consequences of a previously evaluated accident.

Some of the proposed changes represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. The proposed amendment for Dresden and Quad Cities Station's Technical Specifications are based on STS guidelines or later operating BWR plants' NRC accepted changes. Any deviations from STS requirements do not significantly increase the probability or consequences of any previously evaluated accidents for Dresden or Quad Cities Stations. The proposed amendment is consistent with the current safety analyses and has been previously determined to represent sufficient requirements for the assurance and reliability of equipment assumed to operate in the safety analysis, or provide continued assurance that specified parameters remain within their acceptance limits. As such, these changes will not significantly increase the probability or consequences of a previously evaluated accident.

The associated systems related to this proposed amendment are not assumed in any safety analysis to initiate any accident sequence for Dresden or Quad Cities Stations; therefore, the probability of any accident previously evaluated is not increased by the proposed amendment. In addition, the proposed surveillance requirements for the proposed amendments to these systems are generally more prescriptive than the current requirements specified within the Technical Specifications. The additional surveillance requirements improve the reliability and availability of all affected systems and therefore, reduce the consequences of any accident previously evaluated as the probability of the systems related to the TSUP open items outlined within the proposed Technical Specifications performing their intended function is increased by the additional surveillances.

Create the possibility of a new or different kind of accident from any previously evaluated because:

In general, the proposed amendment represents the conversion of current requirements to a more generic format, the addition of requirements which are based on the current safety analysis, and some minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. These changes do not involve revisions to the design of the station. Some of the changes may involve revision in the operation of the station; however, these provide additional restrictions which are in accordance with the current safety analysis, or are to provide for additional testing or surveillances which will not introduce new failure mechanisms beyond those already considered in the current safety analyses.

The proposed amendment for Dresden and Quad Cities Station's Technical Specification is based on STS guidelines or later operating BWR plants' NRC accepted changes. The proposed amendment has been reviewed for acceptability at the Dresden and Quad Cities Nuclear Power Stations considering similarity of system or component design versus the STS or later operating BWRs. Any deviations from STS requirements do not create the possibility of a new or different kind of accident previously evaluated for Dresden or Quad Cities Stations. No new modes of operation are introduced by the proposed changes. Surveillance requirements are changed to reflect improvements in technique, frequency of performance or operating experience at later plants. Proposed changes to action statements in many places add requirements that are not in the present technical specifications. The proposed changes maintain at least the present level of operability. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

The associated systems related to this proposed amendment are not assumed in any safety analysis to initiate any accident sequence for Dresden or Quad Cities Stations. In addition, the proposed surveillance requirements for affected systems associated with the TSUP open items are generally more prescriptive than the current requirements specified within the Technical Specifications; therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

ATTACHMENT D

Involve a significant reduction in the margin of safety because:

In general, the proposed amendment represents the conversion of current requirements to a more generic format, the addition of requirements which are based on the current safety analysis, and some minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. Some of the latter individual items may introduce minor reductions in the margin of safety when compared to the current requirements. However, other individual changes are the adoption of new requirements which will provide significant enhancement of the reliability of the equipment assumed to operate in the safety analysis, or provide enhanced assurance that specified parameters remain with their acceptance limits. These enhancements compensate for the individual minor reductions, such that taken together, the proposed changes will not significantly reduce the margin of safety.

The proposed amendment to the Technical Specifications implements present requirements, or the intent of present requirements in accordance with the guidelines set forth in the STS. Any deviations from STS requirements do not significantly reduce the margin of safety for Dresden or Quad Cities Stations. The proposed changes are intended to improve readability, usability, and the understanding of technical specification requirements while maintaining acceptable levels of safe operation. The proposed changes have been evaluated and found to be acceptable for use at Dresden or Quad Cities based on system design, safety analysis requirements and operational performance. Since the proposed changes are based on NRC accepted provisions at other operating plants that are applicable at Dresden or Quad Cities and maintain necessary levels of system or component reliability, the proposed changes do not involve a significant reduction in the margin of safety.

The proposed amendment for Dresden and Quad Cities Stations will not reduce the availability of systems associated with the TSUP open items when required to mitigate accident conditions; therefore, the proposed changes do not involve a significant reduction in the margin of safety.

ATTACHMENT D

ENVIRONMENTAL ASSESSMENT STATEMENT APPLICABILITY REVIEW

ComEd has evaluated the proposed supplemental amendment against the criteria for the identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.20. It has been determined that the proposed supplemental changes meet the criteria for a categorical exclusion as provided under 10 CFR 51.22 (c)(9). This conclusion has been determined because the supplemental changes requested do not pose significant hazards consideration or do not involve a significant increase in the amounts, and no significant changes in the types, of any effluent that may be released offsite. Additionally, this request does not involve a significant increase in individual or cumulative occupational radiation exposure. Therefore, the Environmental Assessment Statement is not applicable for these supplemental changes.

Commonwealth Edison Company
1400 Opus Place
Downers Grove, IL 60515

ComEd

September 12, 1995

U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attn: Document Control Desk

Subject: Dresden Nuclear Power Station Units 2 and 3
Quad Cities Nuclear Power Station Units 1 and 2
Additional Information; Technical Specification Upgrade Program (TSUP)
Section 3/4.2, "Instrumentation"
NRC Docket Nos. 50-237/249 and 50-254/265

Reference: J.L. Schrage to USNRC letter dated August 4, 1995

In the referenced letter, Commonwealth Edison (ComEd) provided a response to an NRC Staff Request for Additional Information (RAI) concerning Section 3/4.2 ("Instrumentation") of the proposed Technical Specification Upgrade Program (TSUP) for Dresden and Quad Cities Stations. Upon further review, ComEd has identified an administrative error and three typographical errors in the Attachment to the referenced letter.

The administrative error resulted in the omission of four pages from the Attachment (Comparison Matrices B-1, B-2, B-3 and B-4). Enclosure 1 to this letter provides the omitted pages from the referenced letter.

The typographical errors were also part of the Attachment to the referenced letter. Specifically, three of the entries contained in Comparison Matrix A-5 ("Isolation, ECCS, Rod Block Surveillance Requirements") are in error. These three entries are the Channel Calibration comparisons for the Current Technical Specification (CTS) functions of Reactor Low Low Water Level (ECCS Actuation), Steam Line High Flow (HPCI Isolation), and Low Reactor Pressure (HPCI Isolation). Enclosure 2 to this letter provides a revised Comparison Matrix A-5. The modified sections are noted with a bold outline on the attached revision of the Comparison Matrix.

The correction of the typographical error associated with the Calibration comparison for the Reactor Low Low Water Level (ECCS Actuation) instrument also indicates that there is an apparent deviation from Quad Cities CTS requirements. Due to this typographical error, the deviation from CTS was not discussed or justified in the Attachment to the referenced letter. Enclosure 2 also provides a description and justification of the deviation from Quad Cities CTS requirements for the Channel Calibration of the Reactor Low Low Water Level instruments (TSUP Table 4.2.B-1, 1.a, 2.a, and 3.a).

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The two new Open Items identified by the revised Comparison Matrix A-5 [TSUP Channel Calibration for the Steam Line High Flow and Low Reactor Pressure instruments (HPCI Isolation)] will be resolved in the final "Open Item" resolution submittal.

ComEd sincerely apologizes for any inconvenience that this may have caused. If there are any questions concerning this matter, or need for further clarification, please contact this office.

Sincerely,

John L. Schrage
Nuclear Licensing Administrator

Enclosure 1: Omitted Pages from ComEd Response to USNRC RAI, TSUP Section 3/4.2, "Instrumentation"

Enclosure 2: Revised Comparison Matrix A-5, "Isolation, ECCS, Rod Block Surveillance Requirements"

cc: H.B. Miller, Regional Administrator - RIII
J. F. Stang, Project Manager - NRR
R. M. Pulsifer, Project Manager - NRR
M. N. Leach, Senior Resident Inspector - Dresden
C. G. Miller, Senior Resident Inspector - Quad Cities
Office of Nuclear Facility Safety - IDNS

ENCLOSURE 1

Omitted Pages from ComEd Response to USNRC RAI TSUP Section 3/4.2, "Instrumentation"

Comparison Matrix B-1

Dresden CTS Table 3.2.2

Quad Cities Table 3.2-2

TSUP Tables 3.2.B-1, 3.2.D-1, 3.2.I-1

ECCS ACTUATION INSTRUMENTATION

CTS Instrument	TSUP Item No(s).	CTS Appl. Modes	TSUP Modes	CTS Min. Channels per Trip Function	TSUP Min. Channels per Trip Function	CTS Trip Level Setting	TSUP Setpoint	TSUP Functional Unit
Reactor Low Low Water	1.a, 2.a,	Fuel in vessel; Prior to S/U (Q)	1, 2, 3, 4, 5	4	4	$\geq$ 84" above TAF	$\geq$ 84" above TAF	Reactor Vessel Water Level - Low Low
Reactor Low Low Water	3.a, 4.a, 5.a	Fuel in vessel; Press. > 150 psig (HPCI), 90 (ADS - Q), 150 (ADS - D); Prior to S/U (ADS - Q)	1, 2, 3	4	4	$\geq$ 84" above TAF	$\geq$ 84" above TAF	Reactor Vessel Water Level - Low Low
High Drywell Pressure	1.b, 2.b,	Fuel in vessel; Prior to S/U (Q)	1, 2, 3, 4, 5	4	4	$\leq$ 2 psig (D); $\leq$ 2.5 psig (Q)	$\leq$ 2 psig (D); $\leq$ 2.5 psig (Q)	Drywell Pressure - High
High Drywell Pressure	3.b, 4.b, 5.b	Fuel in vessel; Press. > 150 psig (HPCI), 90 (ADS - Q), 150 (ADS - D); Prior to S/U (ADS - Q)	1, 2, 3	4	4	$\leq$ 2 psig (D); $\leq$ 2.5 psig (Q)	$\leq$ 2 psig (D); $\leq$ 2.5 psig (Q)	Drywell Pressure - High
Reactor Low Pressure	1.c, 2.c	Fuel in vessel; Prior to S/U (Q)	1, 2, 3, 4, 5	2	2	$\geq$ 300 psig and $\leq$ 350 psig	$\geq$ 300 psig and $\leq$ 350 psig	Reactor Vessel Pressure - Low (Permissive)
Containment Spray Interlock - 2/3 Core Height	TSUP Table 3.2.I-1, item 2	Fuel in vessel, Prior to S/U (Q), Rx water temp. > 2120F (D)	1, 2, 3	2	2	$\geq$ 2/3 core height	$\geq$ - 48 inches	Reactor Vessel Water Level - Low (Permissive)
Containment Spray Interlock - Containment High Pressure	TSUP Table 3.2.I-1, item 1	Fuel in vessel, Prior to S/U (Q), Rx water temp. > 2120F (D)	1, 2, 3	4	4	$\geq$ 0.5 psig and $\leq$ 1.5 psig	$\geq$ 0.5 psig and $\leq$ 1.5 psig	Drywell Pressure - (Permissive)
Timer Auto blowdown	4.c, 5.c	Fuel in vessel & Press. > 90 psig (Q), 150 (D); Prior to S/U (Q)	1, 2, 3	2	1	$\leq$ 120 seconds	$\leq$ 120 seconds	Initiation Timer (ADS)
LPCI Pump Discharge Pressure (Dresden)	4.e, 4.f, 5.e, 5.f	Fuel in vessel & Press. > 90 psig (Q), 150 (D); Prior to S/U (Q)	1, 2, 3	4	1/pump	$\geq$ 50 psig and $\leq$ 100 psig	$\geq$ 100 psig and $\leq$ 150 psig	CS (LPCI) Pump Discharge Pressure
LPCI Pump Discharge Pressure (Quad Cities)	4.e, 4.f, 5.e, 5.f	Fuel in vessel & Press. > 90 psig (Q), 150 (D); Prior to S/U (Q)	1, 2, 3	4	1/pump	$\geq$ 100 psig and $\leq$ 150 psig	$\geq$ 100 psig and $\leq$ 150 psig	CS (LPCI) Pump Discharge Pressure
Sustained High Reactor Pressure (Dresden only)	Tsup Table 3.2.D-1	Fuel in vessel; Rx. Press. > 150 psig	1, 2, 3 with Rx press. > 150 psig	4	4	$\leq$ 1070 psig for 15 seconds	$\leq$ 1070 psig for $\geq$ 15 seconds	Reactor Vessel Pressure - High
Undervoltage on Emergency Buses (Quad Cities)	6.a	Fuel in vessel; Prior to S/U (Q)	1, 2, 3, 4, 5	2/Bus	2/Bus	3045 $\pm$ 5% volts	3045 $\pm$ 152 volts; decreasing voltage	4.16 kv Emergency Bus Undervoltage (Loss of Voltage)
4 KV Loss of Voltage Emergency Buses (Dresden)	6.a	Fuel in vessel; Prior to S/U (Q)	1, 2, 3, 4, 5	1/Bus	2/Bus	2930 $\pm$ 5% volts	2930 $\pm$ 146 volts; decreasing voltage	4.16 kv Emergency Bus Undervoltage (Loss of Voltage)
Degraded Voltage on 4 KV Emergency Buses (Quad Cities)	6.b	Fuel in vessel; Prior to S/U (Q)	1, 2, 3, 4, 5	2/Bus	2/Bus	3840 volts $\pm$ 2%; 5 min $\pm$ 5% delay; 7 sec $\pm$ 20% secondary delay	$\geq$ 3845 volts (Unit 1); $\geq$ 3833 volts (Unit 2)	4.16 kv Emergency Bus Undervoltage (Degraded Voltage)
Degraded Voltage on 4 KV Emergency Buses (Dresden)	6.b	Fuel in vessel; Prior to S/U (Q)	1, 2, 3, 4, 5	1/Bus	2/Bus	3708 volts $\pm$ 2%; 5 min $\pm$ 5% delay; 7 sec $\pm$ 20% secondary delay	$\geq$ 3784 volts (Unit 2); $\geq$ 3832 volts (Unit 3)	4.16 kv Emergency Bus Undervoltage (Degraded Voltage)

Comparison Matrix B-2
Dresden CTS Table 3.2.2
Quad Cities Table 3.2-2
TSUP Tables 3.2.B-1, 3.2.D-1, 3.2.I-1

ECCS ACTUATION INSTRUMENTATION ACTIONS

CTS Instrument	Applicable CTS ECCS Systems	TSUP Item No(s).	Dresden CTS Action	Quad Cities CTS Action	TSUP Action Number	**TSUP Action** - "With the minimum number of operable channles less than required by the minimum operable channels per Trip Function requirement:"
Reactor Low Low Water	Core Spray, LPCI, ADS	1.a, 2.a, 4.a, 5.a	"If min. channel requirement cannot be met: for one trip system - trip that system; for both trip systems - Immediately initiated orderly shutdown to cold condition."	"If min. channel requirement cannot be met for one or both trip systems, declare ECCS system inop. and follow TS 3.5 or 3.9."	30	... with 1 channel inop - trip the trip system in 1 hour or declare ECCS system inop; With more than 1 channel inop, declare the ECCS system inop.
Reactor Low Low Water	HPCI	3.a	See above	See above	35	... place at least 1 inop. channel in tripped condition in 1 hour or declare HPCI inop.
High Drywell Pressure	Core Spray, LPCI, ADS	1.b, 2.b, 4.b, 5.b	See above	See above	30	... with 1 channel inop - trip the trip system in 1 hour or declare ECCS system inop; With more than 1 channel inop, declare the ECCS system inop.
High Drywell Pressure	HPCI	3.b	See above	See above	35	.... place at least 1 inop. channel in tripped condition in 1 hour or declare HPCI inop.
Reactor Low Pressure	Core Spray, LPCI; (Modes 1, 2, & 3)	1.c, 2.c	See above	See above	31.b	... declare the ECCS system inop.
Reactor Low Pressure	Core Spray, LPCI; (Modes 4 & 5)	1.c, 2.c	See above	See above	32	.... place the inop. channel in tripped condition in 1 hour.
Containment Spray Interlock - 2/3 Core Height	Drywell and Suppression Chamber Cooling	Table 3.2.I-1, item 2	See above	See above	80	... for one trip system, place at least 1 inop. channel in tripped condition in 1 hour or declare containment sprays inop.; for both trip systems, declare containment sprays inop.
Containment Spray Interlock - Containment High Pressure	Drywell and Suppression Chamber Cooling	Table 3.2.I-1, item 1	See above	See above	80	... for one trip system, place at least 1 inop. channel in tripped condition in 1 hour or declare containment sprays inop.; for both trip systems, declare containment sprays inop.
Timer Auto blowdown	ADS	4.c, 5.c	See above	See above	31.a	... declare the ADS system inop.
LPCI Pump Discharge Pressure (Dresden)	ADS	4.e, 4.f, 5.e, 5.f	See above	See above	31.a	... declare the ADS system inop.
LPCI Pump Discharge Pressure (Quad Cities)	ADS	4.e, 4.f, 5.e, 5.f	See above	See above	31.a	... declare the ADS system inop.
Sustained High Reactor Pressure (Dresden only)	Isolation Condenser	Table 3.2.D-1	See above	See above	40	... with 1 channel inop - trip the channel in 1 hour or declare Isocondenser inop; With more than 1 channel inop, declare the Isocondenser inop.
Undervoltage on Emergency Buses	Core Spray, LPCI	6.a	See above	See above	36	.... place at least 1 inop. channel in tripped condition in 1 hour or declare associated EDG inop. and follow 3.9.A or 3.9.B
Degraded Voltage on 4 KV Emergency Buses	Core Spray, LPCI	6.b	See above	See above	36	.... place at least 1 inop. channel in tripped condition in 1 hour or declare associated EDG inop. and follow 3.9.A or 3.9.B

Comparison Matrix B-3

Dresden CTS Table 3.2.2
Quad Cities Table 3.2-2
TSUP Table 3.2.B-1

ECCS ACTUATION INSTRUMENTATION ADDITIONAL FUNCTIONAL UNITS

TSUP ECCS Actuation Function	TSUP Functional Unit	TSUP Applicable Modes	BWR-STS Applicable Modes	TSUP Min. Channels per Trip Function	BWR-STS Min. Channels per Trip Function	TSUP Action	BWR-STS Action	BWR-STS Item No.
Core Spray	(1.d) Core Spray Pump Discharge Flow - Low (Bypass)	1, 2, 3, 4, and 5	1, 2, 3, 4, and 5	1/loop	1/pump	33	33	1.d
LPCI	(2.d) LPCI Pump Discharge Flow - Low (Bypass)	1, 2, 3, 4, and 5	1, 2, 3, 4, and 5	1/loop	1/pump	33	33	2.d
HPCI	(3.c) Condensate Storage Tank Level - Low	1, 2, 3	1, 2, 3	2	2	35	36	3.c
HPCI	(3.d) Suppression Chamber Water Level - High	1, 2, 3	1, 2, 3	2	2	35	36	3.d
HPCI	(3.e) Reactor Vessel Water Level - High Trip	1, 2, 3	1, 2, 3	1 - Dresden 2 - Quad Cities	2	31	31	3.e
HPCI	(3.f) HPCI Pump Discharge Flow - Low (Bypass)	1, 2, 3	1, 2, 3	1	1	33	33	3.f
HPCI	(3.g) Manual Initiation	1, 2, 3	1, 2, 3	1/system	1/system	34	34	3.g
ADS (Trip System A & B)	(4.d, 5.d) Low Low Level Timer	1, 2, 3	1, 2, 3	1	1	31	31	4.f Reactor Vessel Water Level - Low, Level 3 (Permissive)
ADS (Trip System A & B)	(4.e, 5.e) Core Spray Pump Discharge Pressure - High (Permissive)	1, 2, 3	1, 2, 3	1/pump	1/loop	31	31	4.e

Comparison Matrix B-4

Dresden CTS Table 3.2.2
Quad Cities Table 3.2-2
TSUP Table 3.2.B-1, 3.2.I-1

ECCS ACTUATION INSTRUMENTATION TABLE NOTATION

Dresden CTS Note	Quad Cities CTS Note	TSUP Table 3.2.B-1 Note	TSUP Other
1	1	Relocated	3.2.B, Action 2; TSUP Table 3.2.B-1, Actions 30 - 36
2	2	(f)	n/a
3 and *	n/a	Deleted	n/a
4	3	Relocated	TSUP Table 3.2.I-1, note (b)
5	n/a	(h)	n/a
n/a	4	Deleted	n/a
n/a	5	Deleted	n/a
n/a	n/a	(a)	n/a
n/a	n/a	(b)	n/a
n/a	n/a	(c)	n/a
n/a	n/a	(d)	n/a
n/a	n/a	(e)	n/a
n/a	n/a	(g)	n/a
n/a	n/a	(i)	n/a
n/a	n/a	(j)	n/a

ENCLOSURE 2

Revised Comparison Matrix A-5
"Isolation, ECCS, Rod Block Surveillance Requirements"

The following is a revision of Comparison Matrix A-5 which was originally provided in the referenced letter. The modified sections are noted with a bold outline on the attached revision of the Comparison Matrix.

The correction of the typographical error associated with the Calibration comparison for the Reactor Low Low Water Level (ECCS Actuation) instrument also indicates that there is an apparent deviation from Quad Cities CTS requirements. The apparent deviation from Quad Cities CTS requirements for Channel Calibration of the Reactor Low Low Water Level instruments (Quad Cities TSUP Table 4.2.B-1, 1.a, 2.a, and 3.a) should also have been included in the referenced letter (Item M.1.iii) as less restrictive requirements.

The Quad Cities CTS Channel Calibration frequency of "Quarterly" for the Reactor Low Low Water Level instrument has been revised to "Sesquiannual" for TSUP Table 4.2.B-1, items 1.a, 2.a, and 3.a. The TSUP Channel Calibration frequency for the Reactor Low Low Water level instrument for the ADS actuation (TSUP Table 4.2.B-1, item 4.a) maintains the current frequency of "Quarterly." Based upon the retention of the current frequency for TSUP Table 4.2.B-1, item 4.a, the extended frequency for Table 4.2.B-1, items 1.a, 2.a, and 3.a (which utilize the same instrumentation) does not represent a significant reduction in the level or margin of safety.

Comparison Matrix A-5 (Re[illegible] ISOLATION, ECCS, ROD BLOCK SURVEILLANCE REQUIREMENTS

Dresden CTS Table 4.2.1
Quad Cities Table 4.2-1

CTS Function	TSUP Item Nos.	TSUP Function	Channel Check	Channel Functional Test	Channel Calibration
ECCS Instrumentation					
Reactor Low Low Water	Table 4.2.B-1; 1.a, 2.a, 3.a, 4.a	Reactor Vessel Water Level - Low Low	D/Q - Daily; TSUP - S	D/Q - M; TSUP - M	D/Q - Q; D-TSUP - Q, Q-TSUP - E (1.a, 2.a, 3.a); TSUP - Q (4.a)
High Drywell Pressure	Table 4.2.B-1; 1.b, 2.b, 3.b, 4.b	Drywell Pressure - High	D/Q - None; TSUP - N/A	D/Q - M; TSUP - M	D/Q - Q; TSUP - Q
Reactor Low Pressure	Table 4.2.B-1; 1.c, 2.c	Reactor Vessel Pressure - Low (Permissive)	D/Q - None; TSUP - N/A	D/Q - M; TSUP - M	D/Q - Q; TSUP - Q
Containment Spray Interlock - 2/3 core height	Table 4.2.I-1; item 2	Reactor Vessel Water Level - Low (Permissive)	D/Q - None; TSUP - D	D/Q - M; TSUP - M	(Analog Trip Units/Transmitters) D/Q - M/R; TSUP - M/E
Containment Spray Interlock - Containment High Pressure	Table 4.2.I-1; item 1	Reactor Vessel Pressure - Low (Permissive)	D/Q - None; TSUP - N/A	D/Q - M; TSUP - M	D/Q - Q; TSUP - Q
Low Pressure Core Cooling Pump Discharge	Table 4.2.B-1; 4.e, 4.f	CS (LPCI) Pump Discharge Pressure	D/Q - None; TSUP - N/A	D/Q - M; TSUP - M	D/Q - Q; TSUP - Q
Undervoltage Emergency Bus	Table 4.2.B-1; item 5.a	4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	D - Q, Q - None; TSUP - N/A	D/Q - R; TSUP - E	D/Q - R; TSUP - E
Sustained High Reactor Pressure (Dresden only)	Dresden TSUP Table 4.2.D-1	Reactor Vessel Pressure - High	D - None; TSUP - N/A	D - M; TSUP - M	**OPEN ITEM**
Degraded Voltage Emergency Bus	Table 4.2.B-1; item 5.b	4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	D/Q - M; TSUP - N/A	D/Q - R; TSUP - E	D/Q - R; TSUP - E
Rod Blocks					
APRM Downscale	Table 4.2.E-1; 2.c	APRM Downscale	D/Q - None; TSUP - N/A	D/Q - M; TSUP - S/U, M	**D - OPEN ITEM**; Q - Q, TSUP - SA
APRM Flow Variable	Table 4.2.E-1; 2.a.1, 2.a.2	APRM Flow Biased Neutron Flux - High	D/Q - None; TSUP - N/A	D/Q - M; TSUP - S/U, M	D/Q - R; TSUP - SA
APRM upscale (Startup/Hot Standby) - (Dresden only)	Table 4.2.E-1; 2.d	APRM Startup Neutron Flux - High	D - W or D; TSUP - N/A	D - S/U; TSUP - S/U, M	D - S/U & S/D; TSUP - SA
IRM upscale	Table 4.2.E-1; 4.b	IRM Upscale	D - W or D, Q - None; TSUP - N/A	D/Q - S/U; TSUP - S/U, W	D/Q - S/U & S/D; TSUP - SA
IRM downscale	Table 4.2.E-1; 4.d	IRM Downscale	D - W or D, Q - None; TSUP - N/A	D/Q - S/U; TSUP - S/U, W	D/Q - S/U & S/D; TSUP - SA
IRM detector not in Startup position (not fully inserted in the Core)	Table 4.2.E-1; 4.a	IRM Detector not full in	D/Q - None; TSUP - N/A	D/Q - S/U; TSUP - S/U, W	**OPEN ITEM**
RBM Upscale	Table 4.2.E-1; 1.a	Rod Block Monitor Upscale	D/Q - None; TSUP - N/A	D/Q - M; TSUP - S/U, M	D/Q - R; TSUP - Q
RBM Downscale	Table 4.2.E-1; 1.c	Rod Block Monitor Downscale	D/Q - None; TSUP - N/A	D/Q - M; TSUP - S/U, M	D/Q - Q; TSUP - Q
SRM upscale	Table 4.2.E-1; 3.b	SRM Upscale	D/Q - None; TSUP - N/A	D/Q - S/U; TSUP - S/U, W	**D - OPEN ITEM**; Q - S/U & S/D, TSUP - E
SRM detector not in Startup position	**OPEN ITEM**				
SRM downscale (Quad Cities CTS; D & Q TSUP)	**OPEN ITEM**				
High Water Level in scram discharge volume (SDV)	Table 4.2.E-1; 5.a	Scram Discharge Volume Water Level - High	D/Q - None; TSUP - N/A	D/Q - Q; TSUP - Q	D/Q - None; TSUP - N/A
SDV high water level scram trip bypassed (Quad Cities only)	Table 4.2.E-1; 5.b	SDV Switch in Bypass	D/Q - None; TSUP - N/A	**OPEN ITEM**	Q - None; TSUP - N/A
Main Steamline Isolation					
Steam Tunnel High Temperature	Table 4.2.A-1; 3.e	High Temperature Main Steamline Tunnel	D/Q - None; TSUP - N/A	D/Q - R; TSUP - E	D/Q - R; TSUP - E
Steamline High Flow	Table 4.2.A-1; 3.d	High Flow Main Steam Line	D/Q - D; TSUP - S	D/Q - M; TSUP - M	**D - OPEN ITEM**; Q - Q; TSUP - E
Steamline low pressure	Table 4.2.A-1; 3.c	Low Pressure Main Steamline	D/Q - None; TSUP - N/A	D/Q - M; TSUP - M	D/Q - Q; TSUP - Q
Steamline High Radiation	Table 4.2.A-1; 3.b	High Radiation Main Steamline Tunnel	D/Q - D; TSUP - S	D/Q - M; TSUP - M	**OPEN ITEM**
Reactor Low Low Water Level (Quad Cities only)	Table 4.2.A-1; 3.a	Reactor Vessel Water Level - Low	Q - D; TSUP - S	Q - M; TSUP - M	(Analog Trip Units/Transmitters) D/Q - M/R; TSUP - M/E
HPCI Isolation					
Steam Line High Flow	Table 4.2.A-1, item 6.a	Steam Flow - High	D/Q - None; TSUP - N/A	D/Q - M; TSUP - M	(Analog Trip Units/Transmitters) D/Q - M/R; TSUP - **OPEN ITEM**
Steamline Area High Temperatu	Table 4.2.A-1, item 6.c	Area Temperature - High	D/Q - None; TSUP - N/A	D/Q - R; TSUP - E	D/Q - R; TSUP - E
Low Reactor Pressure	Table 4.2.A-1, item 6.b	Reactor Vessel Pressure - Low	D/Q - None; TSUP - N/A	D/Q - M; TSUP - M	(Analog Trip Units/Transmitters) D/Q - M/R; TSUP - **OPEN ITEM**

ISOLATION, ECCS, ROD BLOCK SURVEILLANCE REQUIREMENTS

CTS Function	TSUP Item Nos.	TSUP Function	Channel Check	Channel Functional Test	Channel Calibration
RCIC Isolation (Quad Cities only)					
High Flow RCIC Steamline	Table 4.2.A-1, item 5.a	(Reactor Core Isolation Cooling) Steam Flow - High	Q - None; TSUP - N/A	Q - Q; TSUP - M	Q - Q; TSUP - Q
RCIC Turbine Area High Temperature	Table 4.2.A-1, item 5.c	(Reactor Core Isolation Cooling) Area Temperature - High	Q - None; TSUP - N/A	Q - R; TSUP - E	Q - R; TSUP - E
Low Reactor Pressure	Table 4.2.A-1, item 5.b	Reactor Vessel Pressure - Low	Q - None; TSUP - N/A	Q - Q; TSUP - M	Q - Q; TSUP - Q
Isolation Condenser Isolation (Dresden only)					
High Flow Isolation Condenser Line Steamline Side	Table 4.2.A-1, item 5.a	(Isolation Condenser) Steam Flow - High	D - None; TSUP - N/A	D- M; TSUP - M	D - Q; TSUP - Q
High Flow Isolation Condenser Condensate Return Side	Table 4.2.A-1, item 5.b	(Isolation Condenser) Return Flow - High	D - None; TSUP - N/A	D- M; TSUP - M	D - Q; TSUP - Q
Containment Monitoring (Dresden CTS Table 4.2.1 only; D and Q TSUP)					
Pressure Indicator - -5 in. Hg to +5 psig	**OPEN ITEM** (for TSUP Tables 3/4.2.F-1)				
Pressure Indicator - 5 in. to + 70 in. Hg	Table 4.2.F-1; item 5	Drywell Pressure - Narrow Range	D - None; TSUP - M	D - None; TSUP - N/A	**OPEN ITEM**
Temperature	Table 4.2.F-1; item 7	Drywell Air Temperature	D - D; TSUP - M	D - None; TSUP - N/A	D - R; TSUP - E
Drywell - Torus Differential Pressure	4.7.H	Drywell - Suppression Chamber Differential Pressure	D - None; TSUP - D	D - None; TSUP - None	**OPEN ITEM**
Torus Water Level Indicator - Narrow Range	**DELETED**				
Torus Water Level - 40 in. sight glass	**DELETED**				
Safety/Relief Valve Monitoring (Dresden CTS Table 4.2.1 only; D and Q TSUP)					
Safety/Relief Valve Position Indicator (Acoustic Monitor)	TSUP Table 4.2.F-1, item 10; TSUP 4.6.F.2	Safety/Relief Valve Position Indicators	D - M; TSUP - M	D - R; TSUP - None	**OPEN ITEM**
Safety/Relief Valve Position Indicator (Temperature Monitor)	TSUP Table 4.2.F-1, item 10; TSUP 4.6.F.2	Safety/Relief Valve Position Indicators	D - M; TSUP - M	D - None; TSUP - None	D - 18 months; TSUP - E
Safety Valve Position Indicator (Acoustic Monitor)	TSUP Table 4.2.F-1, item 10; TSUP 4.6.E.1	Safety/Relief Valve Position Indicators	D - M; TSUP - M	D - R; TSUP - None	**OPEN ITEM**
SafetyValve Position Indicator (Temperature Monitor)	TSUP Table 4.2.F-1, item 10; TSUP 4.6.E.1	Safety/Relief Valve Position Indicators	D - M; TSUP - M	D - None; TSUP - None	D - 18 months; TSUP - E
Reactor Building Vent Isolation and SBGT Initiation					
Refueling Floor Radiation Monitors	TSUP Table 4.2.A-1, item 2.d	Refueling Floor Radiation - High	D/Q - D; TSUP - S	D/Q - M; TSUP - M	**D - OPEN ITEM;** Q - Q, TSUP - E
Steam Jet Air Ejector Off-Gas Isolation (Quad Cities CTS Table 4.2-1 only)	**Relocated to ODCM**				
Control Room Ventilation System Isolation (Quad Cities CTS and TSUP only)					
Reactor Low Water Level	Table 4.2.A-1; 1.a, 2.a, (4.b - TSUP RWCU Isolation)	Reactor Vessel Water Level - Low	Q- D; TSUP - S	Q - M; TSUP - M	(Analog Trip Units/Transmitters) D/Q - M/R; TSUP - M/E
Drywell High Pressure	Table 4.2.B-1; 1.b, 2.b, 3.b, 4.b; Table 4.2.A-1, 1.b, 2.b	Drywell Pressure - High	Q- None; TSUP - NA	Q - M; TSUP - M	Q - Q; TSUP - Q
Main Steamline High Flow	Table 4.2.A-1; item 3.d	MSL Flow - High	Q- D; TSUP - S	Q - M; TSUP - M	Q - Q; TSUP - E
Toxic Gas Analyzer	TSUP 4.2.K	Toxic Gas Monitoring	Q- D; TSUP - S	Q - M; TSUP - M	Q - 18 months; TSUP - E

Commonwealth Edison Company
1400 Opus Place
Downers Grove, IL 60515

September 1, 1995

ComEd

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attn: Document Control Desk

Subject: Dresden Nuclear Power Station Units 2 and 3
Application for Amendment to Facility Operating Licenses DPR-19 and DPR-25, Appendix A, Technical Specifications for Technical Specification Upgrade Program
NRC Docket Nos. 50-237 and 50-249

Reference: J. Schrage memo to T. Murley, dated October 2, 1991.

In 1991, Commonwealth Edison (ComEd) initiated a formal program to enhance Quad Cities Station's performance in various aspects of plant operation. Necessary improvements to the Technical Specifications were identified as one of the Station top priority issues. In support of that effort, Quad Cities submitted revised Technical Specifications to the NRC during the course of the year (the referenced letter included Quad Cities' submittal for Section 6.0). To enhance the Quad Cities effort and to improve the Technical Specifications at Dresden Station, ComEd initiated a combined, two-station, Technical Specification Upgrade Program (TSUP) to revise the Dresden Technical Specifications and improve the Quad Cities submittals. This program has been outlined and discussed with members of the NRR staff.

Pursuant to 10 CFR 50.90, ComEd proposes to amend Appendix A, Technical Specification to Facility Operating Licenses DPR-19 and DPR-25. The proposed amendment reflects Commonwealth Edison's efforts to upgrade existing Dresden Station Units 2 and 3 Technical Specification Section 6.0 "Administrative Controls." An overall description of the proposed amendment is also included in the Executive Summary. ComEd will submit a similar proposed amendment for Quad Cities Station Units 1 and 2 under separate cover.

The proposed amendment request is provided as follows:

1. An Executive Summary of the Technical Specification Upgrade Program and the proposed amendment;

2. A description of the proposed amendment;

3. The proposed Technical Specification pages with the requested changes;

4. The existing Technical Specification pages for DPR-19 and DPR-25 (Dresden). To reduce the administrative requirements to process this amendment package, a list of the deleted pages for Dresden Units 2 and 3 are provided; the current versions of existing pages will be provided separately for your staff's information and for comparative purposes;

A001
1/1

5. The technical differences between the existing Dresden Unit 2 and Unit 3 Technical Specifications; and

6. Commonwealth Edison's evaluation pursuant to 10 CFR 50.92(c) and 10 CFR 51.21;

The proposed amendments have been approved by Commonwealth Edison's On-Site and Off-Site Review in accordance with Company procedures.

The Technical Specification Upgrade Program (TSUP) proposes changes to each section of the existing Technical Specifications. As such, Commonwealth Edison requests that the proposed amendments be approved as submitted but to become effective upon completion of the entire project. It is requested that the proposed changes to Section 6.0 be approved prior to October 13, 1995.

To the best of my knowledge and belief, the statements contained are true and correct. In some respects, these statements are not based on my personal knowledge but obtained information furnished by other Commonwealth Edison employees and consultants. Such information has been reviewed in accordance with Company practice and I believe it to be reliable.

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

If there are any comments or questions concerning this submittal, please direct them to this office.

Sincerely,

John L. Schrage
Nuclear Licensing Administrator

OFFICIAL SEAL
MARY JO YACK
NOTARY PUBLIC STATE OF ILLINOIS
MY COMMISSION EXPIRES: 11/29/97

Mary Jo Yack 9-1-95

Attachments:
1. Executive Summary
2. Description of the Proposed Amendment
3. The proposed Technical Specification Pages
4. Listing of Deleted Technical Specification Pages
5. Technical Difference Matrix
6. Significant Hazards Evaluation and Environmental Assessment

cc:
H.B. Miller, Regional Administrator - RIII
J.F. Stang, Project Manager - Dresden
R.M. Pulsifer, Project Manager - Quad Cities
M.N. Leach, Senior Resident Inspector - Dresden
C.G. Miller, Senior Resident Inspector - Quad Cities
Office of Nuclear Facility Safety - IDNS

ATTACHMENT 1

EXECUTIVE SUMMARY

Technical Specification 6.0

"ADMINISTRATIVE CONTROLS"

EXECUTIVE SUMMARY

The Dresden Technical Specification Upgrade Program (TSUP) was conceptualized in response to lessons learned from the Diagnostic Evaluation Team inspection and the frequent need for Technical Specification interpretations. A comparison study of the Standard Technical Specification (STS), later operating plant's Technical Specifications provisions and Quad Cities Technical Specifications was performed prior to the Dresden and Quad Cities TSUP effort. The study identified potential improvements in clarifying requirements and requirements which are no longer consistent with current industry practices.

The TSUP is not intended to be a complete adoption for the STS. Overall, the Dresden custom Technical Specifications provide for the safe operation of the plant and therefore, only an upgrade is deemed necessary.

In response to an NRC recommendation, Quad Cities combined the Unit 1 and Unit 2 Technical Specifications into one document. The Dresden Unit 2 and Unit 3 Technical Specifications will also be combined into one document. To accomplish the combination of the Units' Technical Specification, a comparison of the Unit 2 and Unit 3 Technical Specification was performed to identify any technical differences. The technical differences are identified in the proposed amendment package for each section.

The TSUP goal is to provide a better tool to station personnel to implement their responsibilities and to ensure that Dresden Station is operated in accordance with current industry practices. The improved Technical Specifications provide for enhanced operation of the plant.

The proposed Dresden TSUP Section 6.0 requirements are consistent with those proposed in ComEd's April 24, 1995 submittal. The proposed changes are as follows: 1) deletion of the "Review, Investigative and Audit Functions"; 2) title changes to reflect the reorganization of ComEd's Nuclear Operations Division; 3) miscellaneous administrative and editorial changes.

The proposed specification is adopted from the Byron and Braidwood Technical Specifications. Commonwealth Edison prefers to maintain Section 6.0 consistent among all of the six nuclear stations. The proposed specifications utilized the Byron/Braidwood specifications because they more closely followed the Standard Technical Specifications.

Specification 6.0 has been reordered and new titles have been added based on STS arrangements and nomenclature. Some sections have moved to be consistent with the Byron and Braidwood Technical Specifications.

Current Specifications 6.7, Environmental Qualification and 6.10, Major Change to Radioactive Waste Treatment Systems are deleted in accordance with Standard Technical Specifications. Section 6.7 has been superseded by 10CFR 50.49 and Section 6.10 was deleted through the implementation of Generic Letter 89-01.

ATTACHMENT 2

DESCRIPTION OF CHANGES

Technical Specification 6.0

"ADMINISTRATIVE CONTROLS"

ATTACHMENT 2

DESCRIPTION OF AMENDMENT REQUEST

The changes proposed in this amendment request are made to 1) improve the understanding and usability of the present technical specifications, 2) incorporate technical improvements, and 3) include some provisions from later operating plants.

GENERIC CHANGES

The format of the proposed TSUP specification is adopted from the Byron and Braidwood Technical Specifications. The proposed format changes are to make TS Section 6.0 consistent among all of ComEd's six nuclear stations. The proposed specifications utilized the Byron/Braidwood specifications because they more closely followed the Standard Technical Specifications. Therefore, the proposed specifications are identical to the approved Byron and Braidwood Technical Specifications except where limited by design or station procedural practices or regulatory requirements.

COMPARISON OF CURRENT TECHNICAL SPECIFICATIONS (CTS) TO TSUP AND BASIS OF THE PROPOSED CHANGES

CTS 6.1 Organization, Review, Investigation and Audit

1. CTS 6.1.A.1 is encompassed within TSUP 6.2.A.1. The proposed deletion of the requirement "... or the Management Plan for Nuclear Operations, Section 3 Organizational Authority, Activity; Section 6 Interdepartmental Relationships." is consistent with ComEd's submittal dated April 24, 1995. The Management Plan is no longer maintained, therefore, this reference has been deleted. The Organizational lines of authority and responsibilities will continue to be documented in the QA Topical Report . Maintaining these requirements in the QA Topical Report will ensure that proposed changes to these requirements will receive appropriate regulatory oversight. NRC review of the Quality Assurance Program is governed by 10 CFR 50.54.

2. CTS 6.1.A.2 is encompassed within TSUP 6.2.A.2. The proposed requirements are equivalent to CTS requirements.

3. CTS 6.1.A.3 is encompassed within TSUP 6.2.A.3. The title "Senior Vice President - Nuclear Operations" has been changed to "Chief Nuclear Officer (CNO)" to be consistent with the current corporate management structure at ComEd. The proposed change is consistent with ComEd's submittal dated April 24, 1995.

4. CTS 6.1.A.4 is encompassed within TSUP 6.2.A.5. The proposed requirements are equivalent to CTS requirements.

5. CTS 6.1.B is encompassed within TSUP 6.2.B.5. Minor administrative changes to the titles of key personnel are proposed to be consistent with current plant terminology. "Licensed Senior Operators" has been modified to "senior reactor operators." "Licensed operators" has been modified to "reactor operators." "Health physics personnel" has been modified to "health physicists." "Equipment operators" has been modified to "auxiliary operators." Regarding overtime restrictions, clarification has been added to allow deviations from the guidelines of

Generic Letter 82-12 as long as they are authorized in advance by the Station Manager or his designee, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

6. CTS 6.1.C is encompassed within TSUP 6.2.B. The requirements of CTS Table 6.1.1 specific to the minimum licensed operator staffing levels during CORE ALTERATIONS are not included into TSUP because they are encompassed within 10 CFR 50.54(m)(2)(iv). Per Operating Licenses DPR-19 and DPR-25, Dresden must satisfy the requirements of 10 CFR 50.54(m)(2)(iv). 10 CFR 50.54(m)(2)(iv) specifies that "Each licensee shall have present, during alteration of the core of a nuclear power unit (including fuel loading or transfer), a person holding a senior operator license or a senior operator license limited to fuel handling to directly supervise the activity and, during this time, the licensee shall not assign other duties to this person." As such, Technical Specification requirements for the minimum licensed operator staffing levels during CORE ALTERATIONS are redundant. Therefore, the proposed change administratively relocates the description of these controls and does not relax the plant's obligations to maintain the appropriate licensed operator staffing levels during CORE ALTERATIONS. The proposed change is consistent with ComEd's submittal dated April 24, 1995.

   The requirements of CTS Table 6.1.1 are encompassed within TSUP 6.2.B. 10 CFR 50.54(m)(2)(i) specifies the number of Operators and Senior Operators required per shift and is dependent upon the operating mode of the Units. Per Operating Licenses DPR-19 and DPR-25, Dresden must satisfy the requirements of 10 CFR 50.54(m)(2)(i). For Dresden, with no units operating, one licensed senior reactor operator and two licensed reactor operators are required to be on-shift. With one or both units operating, two licensed senior reactor operators and three licensed reactor operators are required to be on-shift. TSUP 6.2.C provides additional requirements and role clarification for the STA. The proposed change is consistent with the shift manning requirements as discussed in the Improved Standard Technical Specifications (ITS - NUREG-1433, Revision 1). The reference to 10 CFR 50.54(m)(2)(i) within proposed TSUP 6.2.B.3 ensures that the appropriate shift-manning requirements are maintained. Therefore, the proposed changes administratively relocate the description of these controls and does not relax the plant's obligations to maintain the appropriate licensed operator staffing levels on-shift.

7. CTS 6.1.D is encompassed within TSUP 6.3. At Dresden, the "Health Physics Supervisor" title has been changed to the "Radiation Protection Manager." In addition, the position of "Technical Superintendent" no longer exists. The requirement that the individual filling the position of "Site Engineering Manager" meets the requirements for "Technical Manager" as described in Section 4.2.4 of ANSI N18.1 (1971) is redundant to existing requirements for unit staff and has been deleted. The remainder of the proposed change is consistent with ComEd's submittal dated April 24, 1995.

   The specific details regarding the training of Radiation Protection Technicians has not been retained within TSUP 6.3. The requirements specified in ANSI N18.1 should suffice for defining the training requirements for site personnel. The specific procedural details for delineating the training program for personnel is inappropriate for inclusion within the Technical Specifications as this information is more appropriately contained within station procedures, controlled by 10 CFR 50.59.

8. CTS 6.1. E has not been retained within TSUP 6.4. The proposed TSUP changes relocate the requirements for the fire brigade training and other fire protection administrative controls to the Fire Protection Program as described in the plant's UFSAR. Current license condition 3.G for Dresden Unit 3 and license condition 2.E for Dresden Unit 2 provide adequate control of these requirements. This control ensures that any changes made to the site's fire protection program that adversely affect the ability of the plant to achieve and maintain safe shutdown in the event of a fire require NRC staff review and approval. As such, the current license conditions provide an equivalent level of oversight as the current Section 6.0, Administrative Controls and are therefore, redundant. Because, the relocation of these requirements to the UFSAR does not reduce the controls of existing requirements; as such, the proposed change is administrative in nature and does not reduce existing plant fire protection requirements.

9. CTS 6.1.F has not been retained within TSUP 6.4. The proposed training and re-training requirements for site personnel (licensed and unlicensed) are adequately controlled via the provisions of ANSI N18.1 or by the licensing requirements of the individual's licenses. As such, the requirements specified in CTS 6.1.F are redundant and unnecessary for inclusion in the TS.

10. CTS 6.1.G has not been retained within TSUP 6.0. The requirements contained in this section will be relocated to the ComEd Quality Assurance Program Topical Report CE-1-A. The proposed change is consistent with ComEd's submittal dated April 24, 1995.

11. CTS Table 6.1.1 for Dresden is encompassed within TSUP 6.2.B. The CTS requirements for Dresden Table 6.1.1 regarding three Units has not been retained within TSUP 6.2.B. CTS Table 6.1.1 for Dresden is based on Dresden Unit 1 control room manning requirements at a period of time when Dresden Units 1, 2 and 3 shared a common control room. The Unit 2 and Unit 3 control room has since been modified and excludes Unit 1 requirements. As such, the number of required non-licensed operators has been reduced in proposed TSUP 6.2.B.1 to be consistent with industry standards and practices regarding shift manning requirements. The Unit 1 requirements are specified in the Unit 1 Technical Specifications. In addition, the shift manning requirements for both Units defueled is encompassed by the requirements with Units in Mode 4 or 5, as described above.

    Current Technical Specification provisions at Dresden Station in Table 6.1.1 specify that one (1) RAD MEN (Radiation Protection Men) will be in position under all conditions of units with fuel. Current provisions to Dresden Table 6.1.1 [Note (1)] allow staffing levels to be less than the minimum staffing level for a two (2) hour period, if immediate actions are taken to restore the requirements.

    The proposed requirements eliminate the ambiguities associated with the applicable conditions for manning of the Radiation Protection Technician. Current Technical Specification requirements are unclear regarding applicability and corresponding location of fuel within the nuclear units. Current Dresden provisions specify in Table 6.1.1, "UNITS WITH FUEL." It is unclear if the current reference to fuel regarding the unit is applicable when fuel is in the reactor vessel or when fuel is in the reactor vessel and/or spent fuel storage locations. The proposed requirements explicitly clarify that the manning requirements are applicable for the Radiation Protection Technician when fuel is in the reactor, thus eliminating the current ambiguity.

The proposed requirements specified in TSUP 6.2.B.3 are consistent with those specified in the Improved Standard Technical Specifications (ITS - NUREG-1433). In addition, the proposed requirements are consistent with the provisions specified in the LaSalle County, Braidwood, Byron, River Bend, Perry and Hope Creek Technical Specifications.

The proposed requirements enhance guidance given to shift personnel regarding minimum staffing levels and eliminate ambiguities associated with the current Technical Specification requirements; therefore, the proposed changes provide an adequate level of protection for Radiation Protection Technician shift manning when compared to current requirements.

The Shift Manager (SM) position fulfills the requirements in the Dresden CTS for the number of SROs on shift. As such, the proposed TSUP requirements are equivalent to CTS shift manning requirements for SROs.

12. CTS 6.1.H regarding the Fire Protection Program has not been retained within TSUP 6.0. The requirements contained in this section will be relocated to the ComEd Quality Assurance Program Topical Report CE-1-A. This change is consistent with ComEd's submittal dated April 24, 1995.

CTS 6.2 Procedures and Programs

1. CTS 6.2.A, regarding the controls for written procedures is encompassed within TSUP 6.8.A. The proposed requirements are equivalent to CTS requirements.

2. CTS 6.2.B regarding technical review and control of procedures and CTS 6.2.C regarding temporary changes to procedures and has been deleted from TSUP and relocated to administrative controls. Relocation is based on existing regulations and standards that contain these provisions, such that duplication in TSUP is not necessary. The requirements for the establishment, maintenance and implementation of procedures related to activities affecting quality are contained in 10 CFR 50, Appendix B, Criteria II and V; ANSI N18.7-1976; and ANSI N45.2-1971. Changes to the implementing procedures will be controlled by the requirements of 10 CFR 50.59 to ensure that proper reviews affecting safe operation of the plant are performed.

3. CTS 6.2.D has not been retained within TSUP 6.0. The GSEP Manual requirements are encompassed within CTS 6.2.A.4 that specifies that written procedures shall be established, implemented and maintained covering the activities associated with the implementation of the Generating Station Emergency Response Plan. CTS 6.2.A.4 is retained as TSUP 6.8.A.4. In addition, the proposed changes are consistent to the requirements specified in the Byron/Braidwood Technical Specifications.

CTS 6.3 Reportable Event Action

1. CTS 6.3 has not been retained in TSUP 6.0. Requirements regarding promptly reviewing and reporting of reportable events has not been retained in TSUP 6.0 The organization and responsibilities of individuals and functions are adequately described in plant procedures and the Quality Assurance Program. Eliminating repetition of these details from the Technical Specifications will not compromise plant safety. The removal of these items are consistent with

changes addressed in NRC letter from W. T. Russell to Owners Group Chairmen, dated October 25, 1993. In addition, the proposed changes are consistent with the guidance provided in the BWR Improved Standard Technical Specifications, NUREG-1433.

CTS 6.4 Action to be Taken in the Event a Safety Limit is Exceeded

1. CTS 6.4 regarding administrative actions required in the event a safety limit is exceeded are encompassed within TSUP 6.7. CTS 6.4 nomenclature related to the Vice President BWR Operations promptly reporting the event has been replaced with Site Vice President to reflect the current ComEd organizational structure.

   CTS 6.4 regarding the incident report development has been encompassed within TSUP 6.7.A.2. TSUP provides clarification of the reporting vehicle for the event in that it requires an LER be prepared and submitted to the Commission to document the incident. The TSUP elimination of the review reference to Dresden CTS 6.1.G.1.a and 6.1.G.2.b(10) are consistent to those proposed by ComEd in the April 24, 1995 submittal.

   Regarding Safety Limit Actions, the current requirements specifying the immediate shutdown of the reactor has been deleted from Section 6.0 and relocated to TSUP Section 2.0. Previous TSUP submittals for section 2.0 allow a period of 2 hours to bring the unit to a shutdown conditon and then subsequently initiate the appropriate reporting requirements. The proposed TSUP requirements allow a period of time to assess, evaluate and choose the safest course of action. The current requirements may in fact be imprudent because no time to pause and assess the situation is provided. Thus, during an event or transient that threatens a plant safety limit, immediate shutdown of the reactor may introduce additional uncertainty into the event. The proposed changes have been shown by industry experience and precedence to provide reasonable assurance that the reactor coolant system pressure boundary integrity can be maintained within the requirements of the Standard Technical Specification and the Improved Standard Technical Specifications. The small time frame (2 hours) is insignificant with respect to overall plant vulnerability, and prudently allows a reasonable time period to assess a situation in which a safety limit may be approached and thus, the proposed changes are appropriate.

CTS 6.5 Plant Operating Records

1. Requirements contained in CTS 6.5 have not been retained in TSUP. The requirements related to Record Retention can be adequately controlled in the UFSAR and plant procedures, revisions to which are controlled by 10 CFR 50.59. The removal of these items are consistent with changes addressed in NRC letter from W. T. Russell to Owners Group Chairmen, dated October 25, 1993. In addition, the proposed changes are consistent with the guidance provided in the BWR Improved Standard Technical Specifications, NUREG- 1433.

CTS 6.6 Reporting Requirements

1. CTS 6.6.A.1 has been deleted from TSUP. These requirements can be adequately controlled in the UFSAR and plant procedures by 10 CFR 50.59. Eliminating repetition of these details from the Technical Specifications will not compromise plant safety. The removal of these items are consistent with changes addressed in NRC letter from W. T. Russell to Owners Group Chairmen, dated October 25, 1993. In addition, the proposed changes are consistent with the

guidance provided in the BWR Improved Standard Technical Specifications, NUREG-1433.

2. CTS 6.6.A.2 is encompassed within TSUP 6.9.A.2.a. The proposed TSUP requirements are equivalent to CTS requirements.

3. CTS 6.6.A.3 is encompassed within TSUP 6.9.A.5 and the ODCM. The proposed TSUP requirements are equivalent to CTS requirements.

4. CTS 6.6.A.4.a is encompassed within TSUP 6.9.A.6.a. The proposed TSUP requirements are equivalent to CTS requirements.

5. CTS 6.6.A.4.b is encompassed within TSUP 6.9.A.6.b. The proposed TSUP requirements are equivalent to CTS requirements.

6. CTS 6.6.A.4.c is encompassed within TSUP 6.9.A.6.c. The proposed TSUP requirements are equivalent to CTS requirements.

7. CTS 6.6.A.4.d is encompassed within TSUP 6.9.A.6.c. The proposed TSUP requirements are equivalent to CTS requirements.

8. CTS 6.6.B [Reportable Events] has not been retained in TSUP 6.0. The reporting of reportable events requirement is simply a repeat of that required by 10 CFR 50.73, therefore the regulation need not be repeated within the Technical Specifications. Since there is no change in requirements, and the requirements cannot be changed without prior NRC approval, this is considered an administrative change.

9. CTS 6.6.C.1 is encompassed within TSUP 6.9.A.4. The CTS requirements for a Semi-Annual report have been modified to an Annual report. This change is consistent with the final rule for reducing the regulatory burden on nuclear licensees that was published in the Federal Register (FR) on August 31, 1992. The rule change included a revision to 10 CFR 50.36a regarding the frequency for submitting radiological effluent reports. This change is administrative in nature and makes the Technical Specifications consistent with the requirements of 10 CFR 50.36a. The change does not adversely impact the ability to meet applicable regulatory requirements related to liquid and gaseous effluents. The proposed change will eliminate an unnecessary administrative burden without reducing the protection of the public health and safety. Proposed TSUP 6.9.A.4 is consistent with a similar amendment previously approved for Byron and Braidwood Stations (G. Dick letter to D. Farrar, dated February 2, 1995).

10. CTS 6.6.C.2.a(2) is encompassed within TSUP 6.9.A.3 and the ODCM. The proposed TSUP requirements are equivalent to CTS requirements.

11. CTS 6.6.C.2.a(1) is encompassed within TSUP 6.9.A.2.b. The proposed TSUP requirements are consistent with the requirements in the Byron/Braidwood TS. The proposed reporting requirements for Specific Activity in the reactor coolant ensures the appropriate information, consistent to industry practices, is submitted to the Commission.

12. CTS 6.6.C.3 is encompassed within TSUP 6.9.B. The proposed TSUP requirements are equivalent to CTS requirements.

13. CTS Table 6.6.1 has not been retained in TSUP 6.0. One-time reports, which were required five years within unit commercial service date, and upon completion of initial testing, have been deleted from TSUP. The individual requirements for periodic special reports are described within each individual TSUP specification. Requirements pertaining to Radioactive Source Leak Testing reporting have been relocated to TSUP Section 3.8.G, ACTION 2. Requirements pertaining to an NRC report 90 days after completing a Secondary Containment Leak Rate Test has been deleted from TSUP and relocated to administrative controls. The proposed TSUP requirements are consistent with the requirements in the Byron/Braidwood TS.

CTS 6.7 Environmental Qualification

CTS 6.7.A and CTS 6.7.B regarding the Environmental Qualification requirements has not been retained with TSUP 6.0. CTS 6.7, Environmental Qualification (EQ), is being deleted in accordance with 10CFR 50.49, Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants. 10 CFR 50.49 supersedes the current requirements in the Technical Specifications.

CTS 6.8 Offsite Dose Calculation Manual (ODCM)

1. CTS 6.8.A regarding the definition of the ODCM is encompassed within TSUP 1.0, "Definitions," for the ODCM. The TSUP definition for ODCM has been previously approved by the NRC staff (J. Stang letter to D. Farrar, dated February 16, 1995).

   CTS 6.8.A regarding the submittal of the ODCM at the time of RETS to the Commission is superseded by proposed TSUP 6.14.A.3. The CTS 6.8.A requirements are obsolete and are based upon Dresden and Quad Cities' TS submittals in the early 1980's related to the incorporation of the original Radiological Effluents Technical Specifications (R. Bevan letter to D. Farrar [for Quad Cities], dated June 19, 1984). The proposed TSUP 6.14.A.3 requirements are consistent to the guidance provided in Generic Letter 89-01 and are consistent to the Byron/Braidwood Technical Specification requirements.

2. CTS 6.8.B is encompassed within TSUP 6.14.A.1 and TSUP 6.14.A.2. The proposed TSUP 6.14.A.1 and 6.14.A.2 requirements are consistent to the guidance provided in Generic Letter 89-01 and are consistent with the Byron/Braidwood Technical Specification requirements.

CTS 6.9 Process Control Program (PCP)

1. CTS 6.9.A regarding the definition of the PCP is encompassed within TSUP 1.0, "Definitions," for the ODCM. The TSUP definition for PCP has been previously approved by the NRC staff (J. Stang letter to D. Farrar, dated February 16, 1995).

2. The CTS 6.9.B requirements are obsolete and are based upon Dresden and Quad Cities' TS submittals in the early 1980's related to the incorporation of the original Radiological Effluents Technical Specifications (R. Bevan letter to D. Farrar [for Quad Cities], dated June 19, 1984).

The proposed TSUP 6.13.A requirements are consistent with the guidance provided in Generic Letter 89-01 and are consistent to the Byron/Braidwood Technical Specification requirements.

3. CTS 6.9.C is encompassed within TSUP 6.13.A.1 and 6.13.A.2. The proposed TSUP 6.14.A.1 and 6.13.A.2 requirements are consistent to the guidance provided in Generic Letter 89-01 and are consistent to the Byron/Braidwood Technical Specification requirements.

<u>CTS 6.10 Major Changes to Radwaste Treatment Systems</u>

Current Specification 6.10, Major Changes to Radioactive Waste Treatment Systems is being deleted in accordance with Generic Letter 89-01, "Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and Relocation of procedural details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program." The programmatic requirements contained within the current specification are relocated to the Offsite Dose Calculation Manual in accordance with the Generic Letter.

<u>CTS 6.11 Radiation Protection Program</u>

CTS 6.11.1 is encompassed within TSUP 6.11. The proposed TSUP requirements are equivalent to CTS requirements.

<u>CTS 6.12 High Radiation Area</u>

1. CTS 6.12.1 is encompassed within TSUP 6.12.A. TSUP incorporates the definition of HIGH RADIATION AREA as revised in 10 CFR Part 20. TSUP Section 6.12.A describes administrative controls for HIGH RADIATION AREA(s) when dose rates are above 100 mrem/hr at 30 cm (12 in.). The proposed TSUP requirements are equivalent to CTS requirements.

2. CTS 6.12.2 is encompassed within TSUP 6.12.B. TSUP removes the requirement to establish a stay time for personnel entering HIGH RADIATION AREA(s) with dose rates above 1000 mrem/hr at 30 cm (12 in.). TSUP conservatively includes requirements such that persons entering a HIGH RADIATION AREA with dose rates above 1000 mrem/hr at 30 cm (12 in.) to have an alarming radiation monitoring device or to have surveillance and radiation monitoring by a qualified Radiation Protection Technician. This ensures that exposure control is maintained.

   In emergency situations which involve personnel injury or actions taken to prevent major equipment damage, surveillance and radiation monitoring of the work area by a qualified individual may be substituted for routine RWP procedures.

   The proposed TSUP requirements meet the intent of the original CTS requirements.

<u>Miscellaneous New Requirements</u>

1. Specification 6.1, "Responsibility," is a new specification that provides clarification and enhanced guidance regarding the roles and responsibilities of site leadership. The proposed requirements

are consistent with the TS requirements located within the Byron/Braidwood TS.

2. Specification 6.8.B.1 is a new specification for the program Reactor Coolant Sources Outside Primary Containment. The proposed program ensures that leakage from those portions of systems outside primary containment that contain highly radioactive liquid, remain as low as possible. The proposed specification replaces the current license condition for Systems Integrity for DPR-25 (Dresden Unit 3). There is no such license condition in DPR-19 (Dresden Unit 2). The marked-up revised license pages are included in Attachment 4.

3. Specification 6.8.B.2 is a new specification for the program In-Plant Radiation Monitoring. The proposed program ensures the capability to accurately determine the airborne iodine concentrations. The proposed specification replaces the current license condition for Iodine Monitoring for DPR-25 (Dresden Unit 3). There is no such license condition in DPR-19 (Dresden Unit 2). The marked-up revised license page is included in Attachment 4.

4. Specification 6.8.B.3 is a new specification for the program Post Accident Sampling. The proposed program ensures the capability to obtain and analyze reactor coolant, gaseous effluents, and containment atmosphere samples under accident conditions.

5. Specification 6.8.B.4 is a new specification for the Radioactive Effluent Controls Program. The programs ensures that the doses to the members of the public from radioactive effluents will remain as low as reasonably achievable.

SUMMARY AND SCHEDULE

The proposed changes to the Dresden Station Technical Specifications have been reviewed and approved by the Onsite Review in accordance with controlled Station Procedures. Commonwealth Edison has reviewed these proposed amendments in accordance with 10CFR 50.92(c) and determined that no significant hazards consideration exist. This evaluation is documented in Attachment 6. It is requested that the proposed amendment be approved no later than October 13, 1995 and made effective upon completion of the entire Technical Specification Upgrade Program.

ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATIONS

Technical Specification 6.0

"ADMINISTRATIVE CONTROLS"

ATTACHMENT 4

EXISTING TECHNICAL SPECIFICATIONS

Technical Specification 6.0

"ADMINISTRATIVE CONTROLS"

ATTACHMENT 4

DELETION OF CURRENT TECHNICAL SPECIFICATIONS

This technical specification amendment will replace the current section 6.0, Administrative Controls, for the Dresden Unit 2 and Unit 3 Technical Specifications. The specifications are replaced in its entirety with revised pages that combine the Unit 2 and Unit 3 specifications. In addition, the proposed TS changes relocate requirements from the Operating License to TSUP Section 6.0. To reduce the administrative requirements to process this amendment package, a list of the deleted pages for Dresden Units 2 and 3 are provided; the current versions of existing pages will be provided separately for your staff's information and for comparative purposes. In addition, a marked-up version of the revised page from Facility Operating License DPR-25 (which deletes License Conditions 2.I and 2.K) is included in this Attachment.

Delete the following pages:

DPR - 19	DPR - 25	DPR - 19	DPR - 25
6-1	6-1	6-24	6-24
6-2	6-2	6-25	6-25
6-3	6-3	6-26	6-26
6-4	6-4	6-27	--
6-5	6-5	--	--
6-6	6-6	--	--
6-7	6-7	--	--
6-8	6-8	--	--
6-9	6-9	--	--
6-10	6-10	--	--
6-11	6-11	--	--
6-12	6-12	--	--
6-13	6-13	--	--
6-14	6-14	--	--
6-15	6-15	--	--
6-16	6-16	--	--
6-17	6-17	--	--
6-18	6-18	--	--
6-19	6-19	--	--
6-20	6-20	--	--
6-21	6-21	--	--
6-22	6-22	--	--
6-23	6-23	--	--

ATTACHMENT 5

DRESDEN 2/3 DIFFERENCES

Technical Specification 6.0

"<u>ADMINISTRATIVE CONTROLS</u>"

'Deleted'

H. (3) Deleted per Amendment 95.

I. System Integrity

Am. 48
2/06/81

The licensee shall implement a program to reduce leakage from systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

2. Leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

Am. 49
(see 3H)

J. Deleted.

K. Iodine Monitoring

Am. 48
2/06/81

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel;

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

'Deleted'

ATTACHMENT 5

COMPARISON OF DRESDEN UNIT 2 AND UNIT 3 TECHNICAL SPECIFICATIONS FOR THE IDENTIFICATION OF TECHNICAL DIFFERENCES

SECTION 6.0 "ADMINISTRATIVE CONTROLS"

Commonwealth Edison has conducted a comparison review of the Dresden Unit 2 and Unit 3 Technical Specifications to identify any technical differences in support of combining the Technical Specifications into one document. The intent of the review was not to identify any differences in presentation style (e.g. table formats, use of capital letters, etc.), punctuation or spelling errors, but rather to identify areas which the Technical Specifications are technically or administratively different.

The review of Section 6.0 "Administrative Controls" did not reveal any technical differences.

ATTACHMENT 6

SIGNIFICANT HAZARDS CONSIDERATIONS AND ENVIRONMENTAL ASSESSMENT EVALUATION

Technical Specification 6.0

"ADMINISTRATIVE CONTROLS"

ATTACHMENT 6

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Commonwealth Edison has evaluated this proposed amendment and determined that it involves no significant hazards consideration. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility, in accordance with the proposed amendment, would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3) Involve a significant reduction in a margin of safety.

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because:

In general, the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Implementation of these changes will provide increased reliability of equipment assumed to operate in the current safety analysis, or provide continued assurance that specified parameters remain within their acceptance limits, and as such, will not significantly increase the probability or consequences of a previously evaluated accident.

Some of the proposed changes represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. The proposed amendment for Dresden Station's Technical Specification Section 6.0 are based on STS guidelines or later operating plant's NRC accepted changes. Any deviations from STS requirements do not significantly increase the probability or consequences of any previously evaluated accidents for Dresden Station. The proposed amendment is consistent with the current safety analyses and has been previously determined to represent sufficient requirements for the assurance and reliability of equipment assumed to operate in the safety analysis, or provide continued assurance that specified parameters remain within their acceptance limits. As such, these changes will not significantly increase the probability or consequences of a previously evaluated accident.

Create the possibility of a new or different kind of accident from any previously evaluated because:

In general, the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Others represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. These changes do not involve revisions to the design of the station. Some of the changes may involve revision in the operation of the station; however, these provide additional restrictions which are in accordance with the current safety analysis, or are to provide for additional testing or surveillances which will not introduce new failure mechanisms beyond those already considered in the current safety analyses.

The proposed amendment for Dresden Station's Technical Specification Section 6.0 is based on STS guidelines or later operating plants' NRC accepted changes. The proposed amendment has been reviewed for acceptability at the Dresden Nuclear Power Station considering similarity of system or component design versus the STS or later operating plants. Any deviations from STS requirements do not create the possibility of a new or different kind of accident previously evaluated for Dresden Station. No new modes of operation are introduced by the proposed changes. The proposed changes maintain at least the present level of operability. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

Involve a significant reduction in the margin of safety because:

In general, the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Others represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. Some of the later individual items may introduce minor reductions in the margin of safety when compared to the current requirements. However, other individual changes are the adoption of new requirements which will provide significant enhancement of the reliability of the equipment assumed to operate in the safety analysis, or provide enhanced assurance that specified parameters remain with their acceptance limits. These enhancements compensate for the individual minor reductions, such that taken together, the proposed changes will not significantly reduce the margin of safety.

The proposed amendment to Technical Specification Section 6.0 implements present requirements, or the intent of present requirements in accordance with the guidelines set forth in the STS. Any deviations from STS requirements do not significantly reduce the margin of safety for Dresden Station. The proposed changes are intended to improve readability, usability, and the understanding of technical specification requirements while maintaining acceptable levels of safe operation. The proposed changes have been evaluated and found to be acceptable for use at Dresden based on system design, safety analysis requirements and operational performance. Since the proposed changes are based on NRC accepted provisions at other operating plants that are applicable at Dresden and maintain necessary levels of system or component reliability, the proposed changes do not involve a significant reduction in the margin of safety.

ATTACHMENT 6

ENVIRONMENTAL ASSESSMENT

Commonwealth Edison has evaluated the proposed amendment against the criteria for the identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.20. It has been determined that the proposed changes meet the criteria for a categorical exclusion as provided under 10 CFR 51.22 (c)(9). This conclusion has been determined because the changes requested do not pose significant hazards consideration or do not involve a significant increase in the amounts, and no significant changes in the types, of any effluent that may be released offsite. Additionally, this request does not involve a significant increase in individual or cumulative occupational radiation exposure. Therefore, the Environmental Assessment Statement is not applicable for these changes.

ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.A The Station Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.B The Shift Manager shall be responsible for directing and commanding the safe overall operation of the facility under all conditions.

6.2 ORGANIZATION

6.2.A Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for 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 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 departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Quality Assurance Manual.

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 the plant.

3. The Chief Nuclear Officer (CNO) shall have 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 operating pressures.

6.2.B Unit Staff

The unit staff shall include the following:

1. Three non-licensed operators shall be on site at all times.

2. At least one licensed Reactor Operator shall be present in the control room when fuel is in the reactor. In addition, while the unit is in MODE(s) 1, 2, 3 or 4 at least one licensed Senior Reactor Operator shall be present in the control room.

3. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 6.2.B.1 and 6.2.C for a period of time not to exceed two hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

4. A Radiation Protection Technician shall be on site when fuel is in the reactor. The position may be vacant for not more than two hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

5. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g, senior reactor operators, reactor operators, health physicists, auxiliary operators, and key maintenance personnel.

   The amount of overtime worked by unit staff members performing safety-related functions shall be limited in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12). Any deviations from the guidelines of Generic Letter 82-12 shall be authorized in advance by the Station Manager or his designee, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

6. The Operations Manager or Shift Operations Supervisor shall hold a Senior Reactor Operator License.

6.2.C Shift Technical Advisor

The Shift Technical Advisor (STA) shall provide technical advisory support to the Unit Supervisor in the areas of thermal hydraulics, reactor engineering and plant analysis with regard to the safe operation of the facility. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. A single STA may fulfill this function for both units.

6.3 UNIT STAFF QUALIFICATIONS

Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971, "Selection and Training of Nuclear Plant Personnel", dated March 8, 1971, except for the Radiation Protection Manager, who shall meet or exceed the qualifications of the Radiation Protection Manager as specified in Regulatory Guide 1.8, September 1975, and the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and response and analysis of the plant for transients and accidents.

6.4 TRAINING

A retraining and replacement program for the unit staff shall be maintained under the direction of the appropriate on site manager. Training shall be in accordance with ANSI N18.1-1971 and 10 CFR 55 for appropriate designated positions and shall include familiarization with relevant industry operational experience.

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6.7 SAFETY LIMIT VIOLATION

6.7.A The following actions shall be taken in the event a Safety Limit is violated:

1. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Site Vice-President or his designated alternate shall be notified within 24 hours;

2. Within 30 days, a Licensee Event Report (LER) shall be prepared documenting the event pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC.

3. Critical operation of the Unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.A Written procedures shall be established, implemented, and maintained covering the activities referenced below:

1. The applicable procedures recommended in Appendix A, of Regulatory Guide 1.33, Revision 2, February 1978,

2. The Emergency Operating Procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Section 7.1 of Generic Letter No. 82-33,

3. Station Security Plan implementation,

4. Generating Station Emergency Response Plan implementation,

5. PROCESS CONTROL PROGRAM (PCP) implementation,

6. OFFSITE DOSE CALCULATION MANUAL (ODCM) implementation, and

7. Fire Protection Program implementation.

6.8.B The following programs shall be established, implemented, and maintained:

1. Reactor Coolant Sources Outside Primary Containment

   This program provides controls to minimize leakage from those portions of systems outside primary containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include CS, HPCI, LPCI, IC, process sampling, containment monitoring, and standby gas treatment systems. The program shall include the following:

   a. Preventive maintenance and periodic visual inspection requirements, and

   b. Leak test requirements for each system at a frequency of at least once per operating cycle.

2. In-Plant Radiation Monitoring

   This program provides controls which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

   a. Training of personnel,

   b. Procedures for monitoring, and

   c. Provisions for maintenance of sampling and analysis equipment.

3. Post Accident Sampling

   This program provides controls which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and primary containment atmosphere samples under accident conditions. The program shall include the following:

   a. Training of personnel,

   b. Procedures for sampling and analysis,

   c. Provisions for maintenance of sampling and analysis equipment.

4. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC[a][b][c][d][e] from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by station procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

b. Limitations on the instantaneous concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to ten (10) times the concentration values in 10 CFR Part 20, Appendix B, Table 2, Column 2 to 10 CFR Part 20.1001 - 20.2402,

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

d. Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each Unit conforming to Appendix I to 10 CFR Part 50,

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,

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a A MEMBER OF THE PUBLIC shall be an individual in a CONTROLLED or UNRESTRICTED AREA. An individual is not a MEMBER OF THE PUBLIC during any period in which the individual receives an occupational dose.

b The CONTROLLED AREA shall be an area, outside of a RESTRICTED AREA but inside the SITE BOUNDARY, access to which can be limited by the licensee for any reason.

c An UNRESTRICTED AREA shall be any area, access to which is neither limited nor controlled by the licensee.

d RESTRICTED AREA shall be an area, access to which is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. RESTRICTED AREA(s) do not include areas used as residential quarters, but separate rooms in a residential building may be set apart as a RESTRICTED AREA.

e The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

f. Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose conforming to Appendix I to 10 CFR Part 50,

g. Limitations on the dose rate resulting from radioactive materials released in gaseous effluents from the site to areas at or beyond the SITE BOUNDARY shall be limited to the following:

   a) For noble gases: less than or equal to a dose rate of 500 mrem/yr to the whole body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and

   b) For Iodine-131, Iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: less than or equal to a dose rate of 1500 mrem/yr to any organ.

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each Unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

i. Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with halflives greater than 8 days in gaseous effluents released from each Unit conforming to Appendix I to 10 CFR Part 50,

j. Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

6.9 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Regional Administrator of the appropriate Regional Office of the NRC unless otherwise noted.

6.9.A. Routine Reports

1. Deleted

2. Annual Report

   Annual reports covering the activities of the Unit for the previous calendar year, as described in this section shall be submitted prior to March 1 of each year.

   The reports required shall include:

   a. Tabulation of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/year and their associated person rem exposure according to work and job functions, e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter or TLD. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions.

   b. The results of specific activity analysis in which the reactor coolant exceeded the limits of Specification 3.6.J. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the reactor coolant exceeded the radioiodine limit.

3. Annual Radiological Environmental Operating Report

   The Annual Radiological Environmental Operating Report covering the operation of the Unit during the previous calendar year shall be submitted prior to May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

4. Radioactive Effluent Release Report

   The Radioactive Effluent Release Report covering the operation of the facility during the previous calendar year shall be submitted prior to April 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the facility. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

5. Monthly Operating Report

   Routine reports of operating statistics and shutdown experience, including documentation of all challenges to safety valves or safety/relief valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the NRC Regional Office, no later than the 15th of each month following the calendar month covered by the report.

6. CORE OPERATING LIMITS REPORT

   a. Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

      (1) The Control Rod Withdrawal Block Instrumentation for Table 3.2.E-1 of Specification 3.2.E.

      (2) The Average Planar Linear Heat Generation Rate (APLHGR) Limit for Specification 3.11.A.

      (3) The Local Steady State Linear Heat Generation Rate (LHGR) for Specification 3.11.D.

      (4) The Minimum Critical Power Operating Limit (including 20% scram insertion time) for Specification 3.11.C. This includes rated and off-rated flow conditions.

b. The analytical methods used to determine the operating limits shall be those previously reviewed and approved by the NRC in the latest approved revision or supplement of topical reports:

(1) ANF-1125(P)(A), "Critical Power Correlation - ANFB."

(2) ANF-524(P)(A), "ANF Critical Power Methodology for Boiling Water Reactors."

(3) XN-NF-79-71(P)(A), "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors."

(4) XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors."

(5) XN-NF-85-67(P)(A), "Generic Mechanical Design for Exxon Nuclear Jet Pump Boiling Water Reactors Reload Fuel."

(6) XN-NF-81-22(P)(A), "Generic Statistical Uncertainty Analysis Methodology."

(7) ANF-913(P)(A), "CONTRANSA2: A Computer Program for Boiling Water Reactor Transient Analysis."

(8) Commonwealth Edison Company Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods", and associated Supplements on Neutronics Licensing Analyses (Supplement 1) and La Salle County Unit 2 Benchmarking (Supplement 2).

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

6.9.B Special Reports

Special reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report.

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6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.A Pursuant to 10 CFR 20.1601(c), in lieu of the requirements of paragraph 20.1601 of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr at 30 cm (12 in.) shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP)[f] (or equivalent document). Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

1. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

2. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them; or

3. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the RWP (or equivalent document).

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f Health Physics personnel or personnel escorted by health physics personnel shall be exempt from the RWP issuance requirements during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

6.12.B In addition to the requirements of 6.12.A, areas accessible to personnel with radiation levels greater than 1000 mrem/hr at 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates shall require the following:

1. Doors shall be locked to prevent unauthorized entry and shall not prevent individuals from leaving the area. In place of locking the door, continuous, direct or electronic surveillance that is capable of preventing unauthorized entry may be used. The keys shall be maintained under the administrative control of the Shift Manager on duty and/or health physics supervision.

2. Personnel access and exposure control requirements of activities being performed within these areas shall be specified by an approved RWP(or equivalent document).

3. Each person entering the area shall be provided with an alarming radiation monitoring device that continuously integrates the radiation dose rate (such as an electronic dosimeter.) Continuous surveillance and radiation monitoring by a Radiation Protection Technician may be substituted for an alarming dosimeter.

4. During emergency situations which involve personnel injury or actions taken to prevent major equipment damage, continuous surveillance and radiation monitoring of the work area by a qualified individual may be substituted for the routine RWP (or equivalent document).

5. For individual HIGH RADIATION AREAS accessible to personnel with radiation levels of greater than 1000 mrem/h at 30 cm (12 in.) that are located within large areas where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual areas, then such individual areas shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.A Changes to the PCP:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

    a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and,

    b. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

2. Shall become effective after approval of the Station Manager.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.A Changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

    a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and,

    b. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

2. Shall become effective after approval of the Station Manager .

3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Report for the period of the report in which any change to the ODCM was made effective. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

Commonwealth Edison Company
1400 Opus Place
Downers Grove, IL 60515

ComEd

August 4, 1995

U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attn: Document Control Desk

Subject: Dresden Nuclear Power Station Units 2 and 3
Quad Cities Nuclear Power Station Units 1 and 2
Response to NRC Staff Request for Additional Information (RAI)
Regarding the Technical Specification Upgrade Program (TSUP)
Section 3/4.2, "Instrumentation"
NRC Docket Nos. 50-237/249 and 50-254/265

References: (a) J. Stang letter to D. Farrar, dated February 22, 1995.

(b) J. Schrage letter to T. Murley, dated August 30, 1994

(c) P. Piet letter to W. Russell, dated March 14, 1994

In Reference (a), the NRC staff requested additional information from Commonwealth Edison (ComEd) to support the review and approval of ComEd's TSUP project. Regarding TSUP Section 3/4.2, the NRC requested further evaluation by ComEd concerning the comparison of current requirements and the proposed TSUP requirements. ComEd submitted TSUP Section 3/4.2, "Instrumentation," to the NRC staff in the Reference (b) letter. This letter provide's ComEd's response to the NRC staff's RAI for TSUP Section 3/4.2.

The information in this letter provides a comprehensive evaluation between current requirements and those proposed in TSUP. This includes a discussion demonstrating the acceptability of any apparent deviations.

The Attachment to this letter (including Enclosure 1) provides ComEd's response to the Reference (a) NRC staff RAI. ComEd's response to Generic Question No. 2 includes supplemental information regarding proposed TSUP Section 3/4.2, as well as additional information regarding the comparison to current Technical Specification requirements.

In Reference (c), ComEd requested complete approval of TSUP by the NRC staff prior to June 30, 1995 in order to fully implement the project at Dresden Station. In order to most effectively implement TSUP at Dresden Station, ComEd's goal is to complete implementation of TSUP at Dresden during October 1995. The goal for implementation at Quad Cities is February 1996. The proposed TSUP Section 3/4.2 requirements are consistent to, and confirm, the current safety analysis as described in the Dresden and Quad Cities stations' UFSARs. Any changes to the UFSAR necessitated by the approval and implementation of TSUP will be incorporated into the UFSAR, where applicable.

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In order to assist the NRC staff in the review of TSUP Section 3/4.2, Enclosure 2 to this submittal provides marked-up copies of the current Dresden Unit 2 Technical Specifications, the current Quad Cities Unit 1 Technical Specifications, and the BWR Standardized Technical Specifications (STS) Revision 4 (NUREG 0123) for the Reactor Protection System requirements. These mark-ups consist of a cross-reference between current Technical Specification requirements, BWR-STS requirements, and those proposed in TSUP 3/4.2. The mark-ups are not intended to replace or supersede the TSUP pages submitted to the NRC staff in Reference (b). As such, these pages have been stamped "For Information Only."

If there are any questions, please contact this office.

Sincerely,

John L. Schrage
Nuclear Licensing Administrator

OFFICIAL SEAL
MARY JO YACK
NOTARY PUBLIC STATE OF ILLINOIS
MY COMMISSION EXPIRES 11/29/97

Mary Jo Yack 8-4-95

Attachment ComEd Response to NRC RAI; Section 3/4.2, "Instrumentation"

Enclosure 1 ComEd Response to Generic Question No. 1; Revised No Significant Hazards Consideration

Enclosure 2 "Information Only" - Marked up Technical Specification pages

H.B. Miller, Regional Administrator - RIII
J. F. Stang, Project Manager - NRR
R. M. Pulsifer, Project Manager - NRR
M. N. Leach, Senior Resident Inspector - Dresden
C. G. Miller, Senior Resident Inspector - Quad Cities
Office of Nuclear Facility Safety - IDNS

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Commonwealth Edison Company
1400 Opus Place
Downers Grove, IL 60515

July 28, 1995

ComEd

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Dresden Nuclear Power Station Units 2 and 3
Quad Cities Nuclear Power Station Units 1 and 2
Response to NRC Staff Request for Additional Information (RAI)
Regarding the Technical Specification Upgrade Program (TSUP)
Section 3/4.5, "Emergency Core Cooling Systems"
NRC Docket Nos. 50-237/249 and 50-254/265

References: (a) J. Stang letter to D. Farrar, dated February 22, 1995.

(b) P. Piet letter to T. Murley, dated September 17, 1993.

In Reference (a), the NRC staff requested additional information from Commonwealth Edison (ComEd) to support the review and approval of ComEd's TSUP project. Regarding TSUP Section 3/4.5, the NRC requested further evaluation by ComEd concerning the comparison of current requirements and the proposed TSUP requirements. ComEd submitted TSUP Section 3/4.5, "Emergency Core Cooling Systems," to the NRC staff on September 17, 1993 (Reference (b)). The purpose of this letter is to respond to the NRC staff's RAI for TSUP Section 3/4.5 and supplement the information previously provided in the Reference (b) submittal. The information contained in this letter provides a comprehensive evaluation comparing current requirements with those proposed in TSUP and provides a discussion demonstrating the acceptability of any apparent deviations. Other portions of ComEd's response to the RAI regarding other sections of TSUP will be forthcoming under separate cover.

Attachments A and B to this letter provide ComEd's response to NRC staff Generic Question No. 1 (supplemental significant hazards evaluation for TSUP 3/4.5) and Generic Question No. 2. Our response to Generic Question No. 2 includes supplemental information regarding proposed TSUP Section 3/4.5 as well as additional information regarding the comparison to current Technical Specification requirements. Attachment C provides ComEd's response to the NRC staff RAI regarding specific issues for TSUP 3/4.5.

In order to most effectively implement TSUP at Dresden Station, ComEd's goal is to complete implementation of TSUP at Dresden during October, 1995. The goal for implementation at Quad Cities is February, 1996.

It should be noted that the proposed TSUP Section 3/4.5 requirements are consistent with and confirm the current safety analysis as described in the UFSAR. Any changes to the UFSAR necessitated by the approval and implementation of TSUP will be incorporated into the UFSAR, where applicable.

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In order to assist in the review of TSUP Section 3/4.5, Attachment D to this submittal contains marked-up copies of the current Dresden Unit 2 and Quad Cities Unit 2 Technical Specifications. The mark-ups consist of a cross-reference between current Technical Specification requirements against those proposed in TSUP 3/4.5. The mark-ups are not intended to replace or supersede the TSUP pages submitted to the NRC staff in Reference (b). As such, these pages have been stamped "For Information Only." In addition, Attachment E to this submittal contains marked-up copies of Section 3/4.5 of the BWR/4 STS, where applicable. These mark-ups serve as a cross-reference between STS and the proposed TSUP requirements. The mark-ups are not intended to replace or supersede the TSUP pages submitted to the NRC staff in References (b). As such, these pages have been stamped "For Information Only."

If there are any questions, please contact this office.

Sincerely,

Peter L. Piet
Nuclear Licensing Administrator

Attachments:
A. ComEd Response to Generic Question No. 1
B. ComEd Response to Generic Question No. 2
C. ComEd Response to Questions on TSUP 3/4.5
D. Marked-Up Current Technical Specification Pages
E. Marked-Up BWR/4 STS Pages

H. J. Miller, Regional Administrator - RIII
J. F. Stang, Project Manager - NRR
R. M. Pulsifer, Project Manager - NRR
M. N. Leach, Senior Resident Inspector - Dresden
C. G. Miller, Senior Resident Inspector - Quad Cities
Office of Nuclear Facility Safety - IDNS

Signed before me on this 28th day,

of July, 1995.

Mary Jo Yack
Notary Public

OFFICIAL SEAL
MARY JO YACK
NOTARY PUBLIC STATE OF ILLINOIS
MY COMMISSION EXPIRES 11/29/97

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Commonwealth Edison Company
1400 Opus Place
Downers Grove, IL 60515

July 20, 1995

ComEd

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Dresden Nuclear Power Station Units 2 and 3
Quad Cities Nuclear Power Station Units 1 and 2
Response to NRC Staff Request for Additional Information (RAI)
Regarding the Technical Specification Upgrade Program (TSUP)
Section 3/4.7, "Containment Systems"
NRC Docket Nos. 50-237/249 and 50-254/265

References: (a) J. Stang letter to D. Farrar, dated February 22, 1995.

(b) P. Piet letter to T. Murley, dated September 17, 1993.

In Reference (a), the NRC staff requested additional information from Commonwealth Edison (ComEd) to support the review and approval of ComEd's TSUP project. Regarding TSUP Section 3/4.7, the NRC requested further evaluation by ComEd concerning the comparison of current requirements and the proposed TSUP requirements. ComEd submitted TSUP Section 3/4.7, "Containment Systems," to the NRC staff on September 17, 1993 (Reference (b)). The purpose of this letter is to respond to the NRC staff's RAI for TSUP Section 3/4.7 and supplement the information previously provided in the Reference (b) submittals. The information contained in this letter provides a comprehensive evaluation comparing current requirements with those proposed in TSUP and provides a discussion demonstrating the acceptability of any apparent deviations. Other portions of ComEd's response to the RAI regarding other Sections of TSUP will be forthcoming under separate cover.

Attachment A and B to this letter provides ComEd's response to NRC staff Generic Question No. 1 (supplemental significant hazards evaluation for TSUP 3/4.7) and Generic Question No. 2. Our response to Generic Question No. 2 includes supplemental information regarding proposed TSUP Section 3/4.7 as well as additional information regarding the comparison to current Technical Specification requirements. Attachment C provides ComEd's response to the NRC staff RAI regarding specific issues for TSUP 3/4.7.

In order to most effectively implement TSUP at Dresden Station, ComEd's goal is to complete implementation of TSUP at Dresden during October, 1995. The goal for implementation at Quad Cities is February 1996.

It should be noted that the proposed TSUP Section 3/4.7 requirements are consistent with and confirm the current safety analysis as described in the UFSAR. Any changes to the UFSAR necessitated by the approval and implementation of TSUP will be incorporated into the UFSAR, where applicable.

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In order to assist in the review of TSUP Section 3/4.7, Attachment D to this submittal contains marked-up copies of the current Dresden Unit 2 and Quad Cities Unit 2 Technical Specifications. The mark-ups consist of a cross-reference between current Technical Specification requirements and those proposed in TSUP 3/4.7. The mark-ups are not intended to replace or supersede the TSUP pages submitted to the NRC staff in Reference (b). As such, these pages have been stamped "For Information Only." In addition, Attachment E to this submittal contains marked-up copies of Section 3/4.6 of the BWR/4 STS, where applicable. These mark-ups serve as a cross-reference between STS and the proposed TSUP requirements. The mark-ups are not intended to replace or supersede the TSUP pages submitted to the NRC staff in References (b). As such, these pages have been stamped "For Information Only."

If there are any questions, please contact this office.

Sincerely,

Peter L. Piet
Nuclear Licensing Administrator

Attachments:
A. ComEd Response to Generic Question No. 1
B. ComEd Response to Generic Question No. 2
C. ComEd Response to Questions on TSUP 3/4.7
D. Marked-Up Current Technical Specification Pages
E. Marked-Up BWR/4 STS Pages

J. B. Martin, Regional Administrator - RIII
J. F. Stang, Project Manager - NRR
R. M. Pulsifer, Project Manager - NRR
M. N. Leach, Senior Resident Inspector - Dresden
C. G. Miller, Senior Resident Inspector - Quad Cities
Office of Nuclear Facility Safety - IDNS

Signed before me on this 20th day,

of July, 1995.

Mary Jo Yack
Notary Public

OFFICIAL SEAL
MARY JO YACK
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES: 11/29/97

Commonwealth Edison Company
1400 Opus Place
Downers Grove, IL 60515

ComEd

July 19, 1995

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attn: Document Control Desk

Subject: Dresden Nuclear Power Station Units 2 and 3
Quad Cities Nuclear Power Station Units 1 and 2
Supplemental Application for Amendment to Facility Operating Licenses DPR-19, DPR-25, DPR-29 and DPR-30, Appendix A, Technical Specifications
NRC Docket Nos. 50-237/249 and 50-254/265

References: (a) P. Piet letter to T. Murley, dated July 29, 1992.

(b) P. Piet letter to U.S. NRC, dated May 9, 1995.

In the Reference (a) letter, pursuant to 10 CFR 50.90, Commonwealth Edison (ComEd) proposed to amend Appendix A, Technical Specifications to Facility Operating Licenses DPR-19, DPR-25, DPR-29 and DPR-30. The proposed amendment reflected ComEd's efforts to upgrade existing Technical Specifications Sections 1.0, "Definitions," Section 3/4.0, "Applicability," and Section 3/4.3, "Reactivity Controls."

As discussed in Reference (b), current Technical Specification (CTS) 4.3.C.2 for Dresden and Quad Cities Stations requires that all control rods be scram time tested after each refueling outage, and that 50% of the control rods be measured for scram times not more frequently than 16 weeks nor less frequently than 32 weeks. The present requirements are replaced with proposed Surveillance Requirement (SR) 4.3.D.3, which is based on BWR-STS 4.1.3.2.c (NUREG 0123, Draft Rev. 4), and requires at least 10% of the control rods, on a rotating basis, to be scram time tested at least once per 120 days of reactor power operation. In addition, the provisions in Quad Cities CTS 4.3.C.2, which require an annual scram test of all control rods, have not been retained within the proposed TSUP 4.3.D. As previously discussed, the present requirements are replaced with proposed SR 4.3.D.3, which is based on BWR-STS 4.1.3.2.c, and requires at least 10% of the control rods, on a rotating basis, to be scram time tested at least once per 120 days of reactor power operation.

The scram time testing requirement of proposed SR 4.3.D.3 has been proven to be successful within the industry for detecting scram time deterioration at operating BWRs with control rod drive systems similar in design to that of Dresden and Quad Cities. The population of the control rods which are subjected to scram timing will be reduced as a result of adopting the BWR-STS SR for scram timing, thus reducing unnecessary, excessive wear to the CRDs. The large number of significant control rod moves imposes a large, extended power reduction and movement of many more control rods. These result in additional and unnecessary challenges to fuel cladding (thermal

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cycles) and control rod positioning. In addition, the extent and time of the load drop induces a core xenon transient that further complicates reactor recovery, making the surveillance evolution a significant challenge to the plant and reactivity management while adding minimal data to the extensive performance database.

The reduction in the population of control rods which are subjected to scram timing does not have an adverse effect on the Minimum Critical Power Ration (MCPR) Safety Limit, thus the current licensing basis remains unaffected.

CTS 4.3.C.2 also includes provisions to perform an evaluation after completion of control rod drive scram tests. These provisions are deleted from proposed SR 4.3.D.3, since the proposed SRs require, through their performance, evaluations of control rod drive scram tests. Thus, the evaluations will continue to be performed, yet controlled by administrative methods outside of the Technical Specifications. The current requirement to submit the results of the scram time tests in the annual operating report to the NRC staff has also been deleted. This requirement is obsolete and unnecessary for inclusion as a Technical Specification requirement. However, scram time data disposition will continue to be performed, thus the current licensing basis remains unaffected.

ComEd requests expedited approval of the proposed Section 4.3.D.3 from the Reference (a) submittal, outside of the TSUP program, for Dresden and Quad Cities Stations. Proposed TS 4.3.D.3 would replace CTS 4.3.C.2. ComEd is requesting this expedited approval in order to support scheduled CRD scram time testing at Dresden and Quad Cities Station in August 1995.

For Dresden Station, the proposed change will first affect the completion of the Surveillance Requirements for Unit 3. For Dresden Unit 3, 50% of the control rods were scram time tested, at power, on March 23, 1995. This schedule would then require scram time testing of 50% of the Unit 3 control rods prior to reaching 30% power during the restart of Unit 3. The approval of proposed SR 4.3.D.3 (as a replacement for CTS 4.3.C.2) will be required by August 11, 1995, in order to accomplish the transition to the new specification (thus allowing 10% scram time testing of control rods). The implementation of the proposed scram time testing requirement will minimize excessive wear to the CRDs, reduce the extent of the power reduction associated with CRD scram time testing (and the accompanying core xenon transient), and reduce unnecessary challenges to fuel cladding (thermal cycles) and control rod positioning. The minimization of unnecessary challenges to fuel cladding is additionally important in order to minimize the challenges to a known pin-hole fuel leak in the Dresden Unit 3 reactor core. If the proposed amendment is not approved for Dresden Station by August 11, 1995, ComEd will be required to perform 50% testing of Dresden Unit 3 control rods in order to ensure compliance with the Technical Specifications. In either case, ComEd will ensure full compliance with the Technical Specification requirements.

For Quad Cities Station, the proposed change will first affect the completion of the Surveillance Requirements for Unit 1. For Quad Cities Unit 1, all control rods were scram time tested, at power, on January 15, 1995. This establishes a required completion date of August 27, 1995 to have scram time tested 50% of the Quad Cities Unit 1 control rods. The approval of proposed SR 4.3.D.3 (as a replacement for CTS 4.3.C.2) will be required by August 20, 1995, in order to to accomplish the transition to the new specification (thus allowing 10% scram time testing of control rods). The implementation of the proposed scram time testing requirement will minimize excessive wear to the CRDs, reduce the extent of the power reduction associated with CRD scram time testing (and the accompanying core xenon transient), and reduce unnecessary challenges to fuel cladding (thermal cycles) and control rod positioning. If the proposed amendment for Quad Cities Station is not approved by August 20, 1995, ComEd will be required to perform 50% testing of Quad Cities Unit 1 control rods in order to ensure compliance with the Technical Specifications. In either case, ComEd will ensure full compliance with the Technical Specification requirements.

By extracting proposed TS Section 4.3.D.3 from the Reference (a) submittal, the original finding of No Significant Hazards Consideration is unaffected by this supplemental application. The supplemental requirements are equivalent to those specified in the Reference (a) submittal.

This supplemental request is purely schedular in nature and as such, does not change the findings that the proposed supplemental application does not involve a significant increase in the probability or consequences of an accident previously evaluated; or create the possibility of a new or different kind of accident from any accident previously evaluated; or involve a significant reduction in a margin of safety. The NRC staff's original findings and basis for a no significant hazards determination was published in Federal Register, Volume 58, Number 119, on June 23, 1993 (pages 34071-34073) and remains unaffected by ComEd's proposed supplemental request.

Attachment A to this letter provides marked-up Technical Specification pages for Dresden Station (DPR-19 and DPR-25) and Quad Cities Station (DPR-29 and DPR-30) which incorporate the proposed change. Attachment B to this letter provides retyped Technical Specification pages for Dresden and Quad Cities Station.

To the best of my knowledge and belief, the statements contained in this document are true and correct. In some respects these statements are not based on my personal knowledge, but on information furnished by other ComEd employees, contractor employees, and/or consultants. Such information has been reviewed in accordance with company practice, and I believe it to be reliable.

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

If there are any questions concerning this matter, please contact this office.

Respectfully,

John L. Schrage
Nuclear Licensing Administrator

OFFICIAL SEAL
JACQUELINE T EVANS
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES: 12/15/97

Jacqueline T. Evans 7/19/95

Attachments

cc: H. J. Miller, Regional Administrator - RIII
J. F. Stang, Project Manager - NRR
R. M. Pulsifer, Project Manager - NRR
M. N. Leach, Senior Resident Inspector - Dresden
C. G. Miller, Senior Resident Inspector - Quad Cities
Office of Nuclear Facility Safety - IDNS

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ATTACHMENT A

Marked-Up Pages

Dresden Station		Quad Cities Station	
DPR-19	DPR-25	DPR-29	DPR-30
3/4.3-11	3/4.3-11	3.3/4.3-6	3.3/4.3-4
		3.3/4.3-7	3.3/4.3-5

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3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

2. At 16 week intervals, at least 50% of the control rod drives shall be tested as in 4.3.C.1 so that every 32 weeks all of the control rods shall have been tested. Whenever 50% or more of the control rod drives have been tested, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

'INSERT'

3. Following completion of each set of scram testing as described above, the results will be compared against the average scram speed distribution used in the transient analysis to verify the applicability of the current MCPR Operating Limit. Refer to Specification 3.5.L.

D. Control Rod Accumulators

At all reactor operating pressures, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around this rod has a:

1. Inoperable accumulator,

2. Directional control valve electrically disarmed while in a non-fully inserted position.

D. Control Rod Accumulators

Once a shift check the status of the pressure and level alarms for each accumulator.

3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

'INSERT'

D. Control Rod Accumulators

At all reactor operating pressures, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around this rod has a:

1. Inoperable accumulator,
2. Directional control valve electrically disarmed while in a non-fully inserted position.

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

2. At 16 week intervals, at least 50% of the control rod drives shall be tested as in 4.3.C.1 so that every 32 weeks all of the control rods shall have been tested. Whenever 50% or more of the control rod drives have been tested, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

3. Following completion of each set of scram testing as described above, the results will be compared against the average scram speed distribution used in the transient analysis to verify the applicability of the current MCPR Operating Limit. Refer to Specification 3.5.L.

D. Control Rod Accumulators

Once a shift check the status of the pressure and level alarms for each accumulator.

a. Close within 30 seconds after receipt of a signal for control rods to scram, and

b. Open when the scram signal is reset.

C. Scram Insertion Times

1. The average scram insertion time, based on the deenergization of the scram pilot valve solenoids at time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Average Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	3.50

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

% Inserted From Fully Withdrawn	Average Scram Times (sec)
5	0.398
20	0.954
50	2.12
90	3.80

2. The maximum scram insertion time for 90% of any operable control rod shall not exceed 7 seconds.

C. Scram Insertion Times

1. After refueling outage and prior to operation above 30% power, with reactor pressure above 800 psig, all control rods shall be subject to scram-time measurements from the fully withdrawn position.The scram times for single rod scram testing shall be measured without reliance on the control rod drive pumps.

2. All control rod drives shall have experienced scram test measurements each year. Also, 50% of the control rod drives in each quadrant of the reactor core shall be measured for the scram times specified in Specification 3.3.C during the

'INSERT'

interval not more frequently than 16 weeks nor less frequently than 32 weeks. These tests shall be performed with a reactor pressure above 800 psig and may be measured during a reactor scram. Whenever all of the control rod drive scram times have been measured, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained. The results of measurements performed on the control rod drives shall be submitted in the annual operating report to the NRC.

3. If Specification 3.3.C.1 cannot be met, the reactor shall not be made supercritical; if operating, the reactor shall be shut down immediately upon determination that average scram time is deficient.

4. If Specification 3.3.C.2 cannot be met, the deficient control rod shall be considered inoperable, fully inserted into the core, and electrically disarmed.

5. If the overall average of the 20% insertion scram time data generated to date in the current cycle exceeds the limit specified in the CORE OPERATING LIMITS REPORT, the MCPR operating limit must be modified as required by Specification 3.5.K.

3. The cycle cumulative mean scram time for 20% insertion will be determined immediately following the testing required in Specifications 4.3.C.1 and 4.3.C.2 and the MCPR operating limit adjusted, if necessary, as required by Specification 3.5.K.

D. Control Rod Accumulators

At all reactor operating pressures, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around that rod has:

1. an inoperable accumulator,

D. Control Rod Accumulators

Once a shift, check the status of the pressure and level alarms for each accumulator.

c. the operating power level shall be limited so that the MCPR will remain above the MCPR fuel cladding integrity safety limit assuming a single error that results in complete withdrawal of any single operable control rod

C. Scram Insertion Times

1. The average scram insertion time, based on the deenergization of the scram pilot valve solenoids at time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	*Average Scram Insertion Times (sec)*
5	0.375
20	0.900
50	2.00
90	3.50

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

% Inserted From Fully Withdrawn	*Average Scram Times (sec)*
5	0.398
20	0.954
50	2.12
90	3.80

2. The maximum scram insertion time for 90% insertion of any operable control rods shall not exceed 7 seconds.

3. If Specification 3.3.C.1 cannot be met, the reactor shall not be made supercritical; if operating, the reactor shall be shut down immediately upon determination that average scram time is deficient.

4. If Specification 3.3.C.2 cannot be met, the deficient control rod shall be con-

C. Scram Insertion Times

1. After refueling outage and prior to operation above 30% power, with reactor pressure above 800 psig, all control rods shall be subject to scram-time measurements from the fully withdrawn position. The scram times for single rod scram testing shall be measured without reliance on the control rod drive pumps.

'INSERT'

2. All control rod drives shall have experienced scram test measurements each year. Also, 50% of the control rod drives in each quadrant of the reactor core shall be measured for the scram times specified in Specification 3.3.C during the interval not more frequently than 16 weeks nor less frequently than 32 weeks. These tests shall be performed with a reactor pressure above 800 psig and may be measured during a reactor scram. Whenever all of the control rod drive scram times have been measured, an evaluation shall be made to

sidered inoperable, fully inserted into the core, and electrically disarmed.

5. If the overall average of the 20% insertion scram time data generated to date in the current cycle exceeds the limit specified in the CORE OPERATING LIMITS REPORT, the MCPR operating limit must be modified as required by Specification 3.5.K.

D. Control Rod Accumulators

At all reactor operating pressures, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around that rod has:

1. An inoperable accumulator,

2. A directional control valve electrically disarmed while in a nonfully inserted position, or

3. A scram insertion greater than maximum permissible insertion time.

If a control rod with an inoperable accumulator is inserted full-in and its directional control valves are electrically disarmed, it shall not be considered to have an inoperable accumulator, and the rod block associated with that inoperable accumulator may be bypassed.

E. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δ k. If this limit is exceeded, the reactor shall be shutdown until the cause has been determined and corrective actions have been taken. In accordance with Specification 6.6, the NRC shall be notified of this reportable occurrence within 24 hours.

F. Economic Generation Control System

Operation of the unit with the economic generation control system with automatic flow control shall be permissible only in the range of 65% to 100% of rated core flow, with reactor power above 20%.

provide reasonable assurance that proper control rod drive performance is being maintained. The results of measurements performed on the control rod drives shall be submitted in the annual operating report to the NRC.

5. The cycle cumulative mean scram time for 20% insertion will be determined immediately following the testing required in Specifications 4.3.C.1 and 4.3.C.2 and the MCPR operating limit adjusted, if necessary, as required by Specification 3.5.K.

D. Control Rod Accumulators

Once a shift, check the status of the pressure and level alarms for each accumulator.

E. Reactivity Anomalies

During the startup test program and startups following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every equivalent full power month.

F. Economic Generation Control System

Prior to entering EGC and once per shift while operating in EGC, the EGC operating parameters will be reviewed for acceptability.

INSERT

The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than 800 psig and, during single control rod scram time tests, with the control rod drive pumps isolated from the accumulators, for at least 10% of the control rods, on a rotating basis, at least once per 120 days of power operation.

ATTACHMENT B

Revised Technical Specification Pages

Dresden Station		Quad Cities Station	
DPR-19	DPR-25	DPR-29	DPR-30
3/4.3-11	3/4.3-11	3.3/4.3-6	3.3/4.3-4
		3.3/4.3-7	3.3/4.3-5

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3.3 LIMITING CONDITION FOR OPERATION
(Cont'd)

2. The maximum scram insertion time for 90 insertion of any operable control rod shall not exceed 7.00 seconds.

D. Control Rod Accumulators

At all reactor operating pressures, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around this rod has a:

1. Inoperable accumulator,

2. Directional control valve electrically disarmed while in a non-fully Inserted position.

4.3 SURVEILLANCE REQUIREMENT
(Cont'd)

2. The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than 800 psig and, during single control rod scram time tests, with the control rod drive pumps isolated from the accumulators, for at least 10% of the control rods, on a rotating basis, at least once per 120 days of power operation.

3. Following completion of each set of scram testing as described above, the results shall be compared against the average scram speed distribution used in the transient analysis to verify applicability of the current MCPR Operating Limit. Refer to Specification 3.5.L

D. Control Rod Accumulators

Once a shift check the status of the pressure and level alarms for each accumulator.

3.3 LIMITING CONDITION FOR OPERATION
(Cont'd)

2. The maximum scram insertion time for 90 insertion of any operable control rod shall not exceed 7.00 seconds.

D. Control Rod Accumulators

At all reactor operating pressures, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around this rod has a:

1. Inoperable accumulator,

2. Directional control valve electrically disarmed while in a non-fully Inserted position.

4.3 SURVEILLANCE REQUIREMENT
(Cont'd)

2. The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than 800 psig and, during single control rod scram time tests, with the control rod drive pumps isolated from the accumulators, for at least 10% of the control rods, on a rotating basis, at least once per 120 days of power operation.

3. Following completion of each set of scram testing as described above, the results shall be compared against the average scram speed distribution used in the transient analysis to verify applicability of the current MCPR Operating Limit. Refer to Specification 3.5.L

D. Control Rod Accumulators

Once a shift check the status of the pressure and level alarms for each accumulator.

a. Close within 30 seconds after receipt of a signal for control rods to scram, and

b. Open when the scram signal is reset.

C. Scram Insertion Times

1. The average scram insertion time, based on the deenergization of the scram pilot valve solenoids at time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Average Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	3.50

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

% Inserted From Fully Withdrawn	Average Scram Times (sec)
5	0.398
20	0.954
50	2.12
90	3.80

2. The maximum scram insertion time for 90% of any operable control rod shall not exceed 7 seconds.

C. Scram Insertion Times

1. After refueling outage and prior to operation above 30% power, with reactor pressure above 800 psig, all control rods shall be subject to scram-time measurements from the fully withdrawn position. The scram times for single rod scram testing shall be measured without reliance on the control rod drive pumps.

2. The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than 800 psig and, during single control rod scram time tests, with the control rod drive pumps isolated from the accumulators, for at least 10% of the control rods, on a rotating basis, at least once per 120 days of power operation.

3. If Specification 3.3.C.1 cannot be met, the reactor shall not be made supercritical; if operating, the reactor shall be shut down immediately upon determination that average scram time is deficient.

4. If Specification 3.3.C.2 cannot be met, the deficient control rod shall be considered inoperable, fully inserted into the core, and electrically disarmed.

5. If the overall average of the 20% insertion scram time data generated to date in the current cycle exceeds the limit specified in the CORE OPERATING LIMITS REPORT, the MCPR operating limit must be modified as required by Specification 3.5.K.

3. The cycle cumulative mean scram time for 20% insertion will be determined immediately following the testing required in Specifications 4.3.C.1 and 4.3.C.2 and the MCPR operating limit adjusted, if necessary, as required by Specification 3.5.K.

D. Control Rod Accumulators

At all reactor operating pressures, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around that rod has:

1. an inoperable accumulator,

D. Control Rod Accumulators

Once a shift, check the status of the pressure and level alarms for each accumulator.

C. the operating power level shall be limited so that the MCPR will remain above the MCPR fuel cladding integrity safety limit assuming a single error that results in complete withdrawal of any single operable control rod.

C. Scram Insertion Times

1. The average scram insertion time. Based on the deenergization of the scram pilot valve solenoids at time zero of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Average Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	3.50

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

% Inserted From Fully Withdrawn	Average Scram Times (sec)
5	0.398
20	0.954
50	2.12
90	3.80

2. The maximum scram insertion time for 90% insertion of any operable control rods shall not exceed 7 seconds.

3. If Specification 3.3.C.1 cannot be met the reactor shall not be made super-critical: if operating the reactor shall be shut down immediately upon determination that average scram time is deficient.

4. If Specification 3.3.C.2 cannot be met. The deficient control rod shall be con-

C. Scram Insertion Times

1. After refueling outage and prior to operation above 30% power, with reactor pressure above 800 psig, all control rods shall be subject to scram-time measurements from the fully withdrawn position. The scram times for single rod scram testing shall be measured without reliance on the control rod drive pumps.

2. The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than 800 psig and, during single control rod scram time tests, with the control rod drive pumps isolated from the accumulators, for at least 10% of the control rods, on a rotating basis, at least once per 120 days of power operation.

sidered inoperable, fully inserted into the core and electrically disarmed.

5. If the overall average of the 20% insertion scram time data generated to date in the current cycle exceeds the limit specified in the CORE OPERATING LIMITS REPORT, the MCPR operating limit must be modified as required by Specification 3.5.K.

5. The cycle cumulative mean scram time for 20% insertion will be determined immediately following the testing required in Specifications 4.3.C.1 and 4.3.C.2 and the MCPR operating limit adjusted, if necessary, as required by Specification 3.5.K.

D. Control Rod Accumulators

At all reactor operating pressures, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around that rod has:

1. An inoperable accumulator,

2. A directional control valve electrically disarmed while in a nonfully inserted position, or

3. A scram insertion greater than maximum permissible insertion time.

If a control rod with an inoperable accumulator is inserted full-in and its directional control valves are electrically disarmed, it shall not be considered to have an inoperable accumulator, and the rod block associated with that inoperable accumulator may be bypassed.

D. Control Rod Accumulators

Once a shift, check the status of the pressure and level alarms for each accumulator.

E. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δ k. If this limit is exceeded, the reactor shall be shutdown until the cause has been determined and corrective actions have been taken. In accordance with Specification 6.6, the NRC shall be notified of this reportable occurrence within 24 hours.

E. Reactivity Anomalies

During the startup test program and startups following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every equivalent full power month.

F. Economic Generation Control System

Operation of the unit with the economic generation control system with automatic flow control shall be permissible only in the range of 65% to 100% of rated core flow, with reactor power above 20%.

F. Economic Generation Control System

Prior to entering EGC and once per shift while operating in EGC, the EGC operating parameters will be reviewed for acceptability.

Commonwealth Edison Company
1400 Opus Place
Downers Grove, IL 60515

June 30, 1995

ComEd

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Dresden Nuclear Power Station Units 2 and 3
Quad Cities Nuclear Power Station Units 1 and 2
Response to NRC Staff Request for Additional Information (RAI)
Regarding the Technical Specification Upgrade Program (TSUP)
Section 3/4.6, "Primary System Boundary"
NRC Docket Nos. 50-237/249 and 50-254/265

References: (a) J. Stang letter to D. Farrar, dated February 22, 1995.

(b) P. Piet letter to T. Murley, dated September 17, 1993.

(c) J. Stang letter to D. Farrar, dated June 13, 1995.

In Reference (a), the NRC staff requested additional information from Commonwealth Edison (ComEd) to support the review and approval of ComEd's TSUP project. Regarding TSUP Section 3/4.6, the NRC requested further evaluation by ComEd concerning the comparison of current requirements and the proposed TSUP requirements. ComEd submitted TSUP Section 3/4.6, "Primary System Boundary," to the NRC staff on September 17, 1993 (Reference (b)). The purpose of this letter is to respond to the NRC staff's RAI for TSUP Section 3/4.6 and supplement the information previously provided in the Reference (b) submittals. The information provided in this letter provides a comprehensive evaluation between current requirements and those proposed in TSUP and provides a discussion demonstrating the acceptability of any apparent deviations. Other portions of ComEd's response to the RAI regarding other Sections of TSUP will be forthcoming under separate cover.

Attachments A and B to this letter provide ComEd's response to NRC staff Generic Question No. 1 (supplemental significant hazards evaluation for TSUP 3/4.6) and Generic Question No. 2. Our response to Generic Question No. 2 includes supplemental information regarding proposed TSUP Section 3/4.6 as well as additional information regarding the comparison to current Technical Specification requirements. Attachment C provides ComEd's response to the NRC staff RAI regarding specific issues for TSUP 3/4.6.

In Section 3.8 of Reference (c), the NRC staff listed as an open item the relocation of current Technical Specification 2.2.B to proposed TSUP 3.6.F. Proposed TSUP 3.6.F is fully discussed herein. In order to most effectively implement TSUP at Dresden Station, ComEd's goal is to complete implementation of TSUP at Dresden during October, 1995. The goal for implementation at Quad Cities is February 1996.

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It should be noted that the proposed TSUP Section 3/4.6 requirements are consistent with and confirm the current safety analysis as described in the UFSAR. Any changes to the UFSAR necessitated by the approval and implementation of TSUP will be incorporated into the UFSAR, where applicable.

In order to assist in the review of TSUP Section 3/4.6, Attachment D to this submittal contains marked-up copies of the current Dresden Unit 2 and Quad Cities Unit 2 Technical Specifications. The mark-ups consist of a cross-reference between current Technical Specification requirements and those proposed in TSUP 3/4.6. The mark-ups are not intended to replace or supersede the TSUP pages submitted to the NRC staff in Reference (b). As such, these pages have been stamped "For Information Only." In addition, Attachment E to this submittal contains marked-up copies of Section 3/4.4 of the BWR/4 STS, where applicable. These mark-ups serve as a cross-reference between STS and the proposed TSUP requirements. The mark-ups are not intended to replace or supersede the TSUP pages submitted to the NRC staff in References (b). As such, these pages have been stamped "For Information Only."

If there are any questions, please contact this office.

Sincerely,

Peter L. Piet
Peter L. Piet
Nuclear Licensing Administrator

Attachments:
- A. ComEd Response to Generic Question No. 1
- B. ComEd Response to Generic Question No. 2
- C. ComEd Response to Questions on TSUP 3/4.6
- D. Marked-Up Current Technical Specification Pages
- E. Marked-Up BWR/4 STS Pages

J. B. Martin, Regional Administrator - RIII
D.M. Skay, Project Manager - NRR
J. F. Stang, Project Manager - NRR
R. M. Pulsifer, Project Manager - NRR
M. N. Leach, Senior Resident Inspector - Dresden
C. G. Miller, Senior Resident Inspector - Quad Cities
Office of Nuclear Facility Safety - IDNS

Signed before me on this 30th day,

of June, 1995.

Mary Jo Yack
Notary Public

OFFICIAL SEAL
MARY JO YACK
NOTARY PUBLIC STATE OF ILLINOIS
MY COMMISSION EXPIRES 11/29/97

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Commonwealth Edison Company
1400 Opus Place
Downers Grove, IL 60515

ComEd

June 29, 1995

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Dresden Nuclear Power Station Units 2 and 3
Partial Implementation of Technical Specification Amendments 134 and 128
NRC Docket Nos. 50-237 and 50-249

References: (a) J. F. Stang to D. L. Farrar letter dated June 13, 1995.

(b) Teleconference between USNRC (J. Stang) and ComEd (P. Piet, P. Holland), dated June 14, 1995.

In Reference (a), the NRC staff issued Amendments 134 and 128 to Appendix A (Technical Specifications) of Facility Operating Licenses DPR-19 and DPR-25 (Dresden Nuclear Power Station Unit 2 and Unit 3). Reference (a) noted that the license amendment was effective immediately, to be implemented no later than December 31, 1995.

During the Reference (b) teleconference, ComEd described the additional reactor power changes that they must implement for Dresden Unit 2 and Unit 3 to ensure that sufficient margin exists from the Condenser Low Vacuum SCRAM setpoint to allow for Circulating Water reversal through the main condenser. Circulating water is reversed through the main condenser weekly to mitigate condenser tube fouling and improve thermal heat transfer. The circulating water is reversed typically after a small power decrease to perform weekly control rod drive exercising in accordance with the Technical Specifications. Increasing circulating water temperatures (due to increasing outside ambient temperature) require an additional reduction in power to ensure sufficient margin to the SCRAM setpoint. The additional power reduction represents an unnecessary challenge to a known pin-hole fuel leak in the Dresden Unit 3 reactor core.

In order to avoid unnecessary cycling of Dresden Unit 3 which is required to perform condenser circulating water reversals, by date of this letter, ComEd will implement the new setpoint for the Condenser Low Vacuum SCRAM approved in Technical Specification Upgrade Program (Section 2.0, Safety Limits and Limiting Safety System Settings) Amendments 134 and 128. In order to appropriately control the implementation of the revised setpoints, the attachment to this letter provides revised versions (in the current format) of the current Technical Specification pages that include the revised settings. ComEd will implement the remainder of Amendments 134 and 128 for Dresden Station during the full implementation of the Technical Specification Upgrade Program.

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The new Condenser Low Vacuum SCRAM setpoint will be adopted in both the Dresden Unit 2 and Unit 3 Technical Specifications. Although Unit 2 does not contain any known fuel defects, circulating water flow reversal is performed weekly and often an additional power decrease is required to ensure adequate margin to the SCRAM setpoint exists. In addition, to avoid any unnecessary confusion to site Operating personnel, the setpoint will be consistent between both Dresden Unit 2 and Unit 3.

If there are any questions regarding this matter, please contact this office.

Respectfully,

Peter L. Piet
Nuclear Licensing Administrator

cc: J. B. Martin, Regional Administrator - RIII
J. F. Stang, Project Manager - NRR
M. N. Leach, Senior Resident Inspector - Dresden
Office of Nuclear Facility Safety - IDNS

Attachment: Unit 2 TS Pages 3/4.1-5 and B 3/4.13
Unit 3 TS Pages 3/4.1-5 and B 3/4.13

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ATTACHMENT

Revised Current Technical Specification Ppages

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TABLE 3.1.1
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Minimum Number Operable Inst. Channels per Trip (1) System	Trip Function	Trip Level Setting	Modes in which Function Must be Operable			Action*
			Refuel (6)	Startup/Hot Standby	Run	
1	Mode Switch in Shutdown		X	X	X	A
1	Manual Scram		X	X	X	A
	IRM					
3	High Flux	(LT/E) 120/125 of Full Scale	X	X	N/A	A
3	Inoperative		X	X	N/A	A
	APRM					
2	High Flux	Specification 2.1.A.1	X	X(8)	X	A or B
2	Inoperative **		X	X(8)	X	A or B
2	High Flux (15% Scram)	Specification 2.1.A.2	X	X	N/A	A
2	High Reactor Pressure	(LT/E) 1060 psig	X(10)	X	X	A
2	High Drywell Pressure	(LT/E) 2 psig	X(7), X(9)	X(7), (9)	X(9)	A
2	Reactor Low Water Level	(GT/E) 1 inch***	X	X	X	A
2 (Per Bank)	High Water Level in Scram Discharge Volume (Thermal and dP Switch)	(LT/E) 40 inches above bottom of the Instrument Volume	X(2)	X	X	A or D
2	Turbine Condenser Low Vacuum	(GT/E) 21 in. Hg Vacuum	X(3)	X(3)	X	A or C
2	Main Steam Line High Radiation	(LT/E) 3 X Normal Full Power Background	X	X	X(11)	A or C
4(5)	Main Steam Line Isolation valve Closure	(LT/E) 10% Valve Closure	X(3)	X(3)	X	A or C
2	Generator Load Rejection, turbine control valve trip system oil pressure low	(GT/E) 460 psig****	X(4)	X(4)	X(4)	A or C
2	Turbine Stop Valve Closure	(LT/E) 10% Valve Closure	X(4)	X(4)	X(4)	A or C
2	Turbine Control - Loss of Control Oil Pressure	(GT/E) 900 psig	X(4)	X(4)	X(4)	A or C

Notes: (LT/E) = Less than or equal to.
(GT/E) = Greater than or equal to.
(Notes continue on next two pages)

3.1 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

Loss of condenser vacuum occurs when the condenser can no longer handle the heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. To prevent the clad safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure. The turbine stop valve closure scram function alone is adequate to prevent the clad safety limit from being exceeded in the event of a turbine trip transient with bypass closure. (Ref. Section 4.4.3 SAR) The condenser low vacuum scram is a backup to the stop valve closure scram and causes a scram before the stop valves are closed and thus the resulting transient is less severe. Scram occurs at 21" Hg vacuum, stop valve closure occurs at 20" Hg vacuum, and bypass closure at 7" Hg vacuum.

High radiation levels in the main steam line tunnel above that due to the normal nitrogen and oxygen radioactivity is an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds 3 times full power background for all condition except for greater than 20% power with hydrogen being injected during which the Main Steam Line trip setting is less than or equal to 3 times full power background with hydrogen addition (See Note 15 of Table 3.1.1). The purpose of this scram is to reduce the source of such radiation to the extent necessary to prevent excessive turbine contamination. Discharge of excessive amounts of radioactivity to the site environs is prevented by the air ejector off-gas monitors which cause an isolation of the main condenser off-gas line provided the limit specified in Specification 3.8 is exceeded.

The main steam line isolation valve closure scram is set to scram when the isolation valves are 10% closed from full open. This scram anticipates the pressure and flux transient, which would occur when the valves close. By scramming at this setting, the resultant transient is insignificant.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. (Ref. Section 7.7.1.2 SAR).

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

TABLE 3.1.1
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Minimum Number Operable Inst. channels per Trip (1) System	Trip Function	Trip Level Setting	Modes in which Function Must be Operable: Refuel (6)	Startup/Hot Standby	Run	Action*
1	Mode Switch in Shutdown		X	X	X	A
1	Manual Scram		X	X	X	A
	IRM					
3	High Flux	(LT/E) 120/125 of Full Scale	X	X	N/A	A
3	Inoperative		X	X	N/A	A
	APRM					
2	High Flux	Specification 2.1.A.1	X	X(8)	X	A or B
2	Inoperative **		X	X(8)	X	A or B
2	High Flux (15% Scram)	Specification 2.1.A.2	X	X	N/A	A
2	High Reactor Pressure	(LT/E) 1060 psig	X(10)	X	X	A
2	High Drywell Pressure	(LT/E) 2 psig	X(7), X(9)	X(7), (9)	X(9)	A
2	Reactor Low Water Level	(GT/E) 1 inch***	X	X	X	A
2 (Per Bank)	High Water Level In Scram Discharge Volume (Float and dP Switch)	(LT/E) 37.25 inches above bottom of the Instrument Volume	X(2)	X	X	A or D
2	Turbine Condenser Low Vacuum	(GT/E) 21 in. Hg Vacuum	X(3)	X(3)	X	A or C
2	Main Steam Line High Radiation	(LT/E) 3 X Normal Full Power Background	X	X	X	A or C
4(5)	Main Steam Line Isolation Valve Closure	(LT/E) 10% Valve Closure	X(3)	X(3)	X	A or C
2	Generator Load Rejection, turbine control valve trip system oil pressure low	(GT/E) 460 psig****	X(4)	X(4)	X(4)	A or C
2	Turbine Stop Valve Closure	(LT/E) 10% Valve Closure	X(4)	X(4)	X(4)	A or C
2	Turbine Control - Loss of Control Oil Pressure	(GT/E) 900 psig	X(4)	X(4)	X(4)	A or C

Notes: (LT/E) = Less than or equal to.
(GT/E) = Greater than or equal to.
(Notes continue on next two pages)

3.1 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or the amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function properly.

Loss of condenser vacuum occurs when the condenser can no longer handle the heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. To prevent the clad safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure. The turbine stop valve closure scram function alone is adequate to prevent the clad safety limit from being exceeded in the event of a turbine trip transient with bypass closure. The condenser low vacuum scram is a backup to the stop valve closure scram and causes a scram before the stop valves are closed and thus the resulting transient is less severe. Scram occurs at 21" Hg vacuum, stop valve closure occurs at 20" Hg vacuum, and bypass closure at 7" Hg vacuum.

High radiation levels in the main steam line tunnel above that due to the normal nitrogen and oxygen radioactivity is an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds three times normal background. The purpose of this scram is to reduce the source of such radiation to the extent necessary to prevent excessive turbine contamination. Discharge of excessive amounts of radioactivity to the site environs is prevented by the air ejector off gas monitors which cause an isolation of the main condenser offgas line provided the limit specified in Specification 3.8 is exceeded.

The main steam line isolation valve closure scram is set to scram when the isolation valves are 10% closed from full open. This scram anticipates the pressure and flux transient, which would occur when the valves close. By scramming at this setting, the resultant transient is insignificant.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. (Ref. Section 7.7.1.2 SAR).

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

July 25, 1996

Ms. Irene Johnson, Acting Manager
Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 OPUS Place, Suite 500
Downers Grove, IL 60515

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION CONCERNING NITROGEN CONTAINMENT ATMOSPHERIC DILUTION SYSTEM COMPLIANCE WITH 10 CFR 50.44 AT DRESDEN AND QUAD CITIES NUCLEAR POWER STATIONS (TAC NOS. M94843, M94844, M94845 AND M94846)

Dear Ms. Johnson:

In a letter dated February 16, 1996, Commonwealth Edison Company informed the staff that you plan on using the purge and vent strategy versus repressurization/purge strategy for primary containment hydrogen control at both Dresden and Quad Cities. A response to the enclosed Request for Additional Information (RAI) is needed in order to complete our review of the acceptability of this change.

Sincerely,

Original signed by
M. David Lynch for:

Robert M. Pulsifer, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-237, 50-249,
50-254, 50-265

Enclosure: RAI

cc w/encl: see next page

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I. Johnson
Commonwealth Edison Company

Dresden Nuclear Power Station
Unit Nos. 2 and 3
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Unit Nos. 1 and 2

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1400 Opus Place, Suite 400
Downers Grove, Illinois 60515

REQUEST FOR ADDITIONAL INFORMATION
PRIMARY CONTAINMENT HYDROGEN CONTROL
AT DRESDEN AND QUAD CITIES

The staff provided an evaluation dated September 12, 1988, of the General Electric Topical Report NEDO-31331, "Emergency Procedure Guidelines, Revision 4," March 1987 to the BWR Owners Group. This evaluation provided guidance for the use of the Purge and Vent Strategy in conjunction with the Nitrogen Containment Atmospheric Dilution System for design basis hydrogen control.

1) In the SER that approved the Emergency Procedure Guidelines (EPG), the staff's stated goal is to limit venting to a "last resort" action. The major staff concern has centered on the appropriate containment pressure for venting. As a result, the venting pressure should be established as high as reasonably achievable. If the primary containment pressure limit (PCPL) is less than the design pressure, the licensee must submit justification which the staff will evaluate on a case by case basis. Accordingly, a reasonable effort should be made by each licensee to increase PCPL as high as practical; e.g., perform adjustments to the pneumatic operating pressure of the SRVs, and consider improving vent valve operability. Provide justification for your approach. How does the PCPL compare to the design pressure? Which of the four criteria contained in the staff evaluation cited above, limit the PCPL?

2) What impact did the change in methodology have on the time to manually initiate nitrogen dilution, maximum required injection flow rate and steady state flow rate?

3) The first step of the PC/H section of the EPGs requires venting/purging, whenever either the suppression chamber or drywell reaches the minimum detectable hydrogen concentration, provided that the offsite radioactivity release rate is expected to remain below the offsite Technical Specification value of the Limiting Condition for Operation (LCO) for the release rate. The staff concluded in its SER that operators should have detailed guidance when conditions dictate removal of hydrogen using a purge and vent strategy and that sufficient safeguards should be established to preclude this action from being implemented during an emergency situation. Identify and provide a summary of the primary containment venting procedure, lineups and valve operations.

4) What is the containment pressure profile versus time? The profile should show the initiation and duration of the vent cycle. What volume of containment atmosphere is released during the cycle? What is the maximum allowable purge flow without repressurizing containment?

5) Do plant-specific procedures exist for analyzing a primary containment air sample in support of Step PC/H-1 in the EPGs? If so, identify and summarize these procedures.

ENCLOSURE

50-237, 249
50-373, 374
50-254, 265

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 14, 1996

Mr. D. L. Farrar
Manager, Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: STAFF REVIEW OF MODIFICATIONS TO REVISION 4 OF THE BWR EMERGENCY PROCEDURE GUIDELINES

Dear Mr. Farrar:

The staff has issued its safety evaluation (SE) on the recent BWROG-proposed modifications to the BWR Emergency Procedure Guidelines. The staff is providing this information to ensure that licensees are aware of the conclusions of the staff's review. Both the staff and the Advisory Committee for Reactor Safeguards (ACRS) agree that for BWRs, injecting standby liquid control through a standpipe below the core, maintenance of level above top-of-active fuel (TAF) is the superior water control strategy in an anticipated transient without scram (ATWS) event. The staff recommends a level around TAF +5 feet (1.52 m), or as high as possible while still maintaining the level at least 2 feet (0.61 m) below the feedwater sparger. Although control at any level between the minimum steam cooling water level and two feet below the feedwater sparger was found to be acceptable, both the staff and ACRS urge that a high-water-level control strategy be adopted. Additional details are provided in the enclosed SE.

You should also note the staff's position on bypassing the Main Steam Isolation Valve (MSIV) high radiation closure interlock during ATWS. The staff agrees with the BWROG's qualitative arguments that keeping the MSIVs open significantly reduces containment loading and makes level control much simpler. However, the acceptability of this change is conditional on a plant-specific evaluation by each licensee to assure that, in the event of gross

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fuel failures, consideration has been given to such items as equipment accessibility, potential off-site radiological doses, and the appropriate time to manually close the MSIVs.

Sincerely,

John F. Stang

John F. Stang, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-237, 50-249, 50-373,
50-374, 50-254, 50-265

Enclosure: Safety Evaluation

cc w/encl: See next page

D. L. Farrar
Commonwealth Edison Company

Dresden Nuclear Power Station
Unit Nos. 2 and 3
Quad Cities Nuclear Power Station
Unit Nos. 1 and 2

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fuel failures, consideration has been given to such items as equipment accessibility, potential off-site radiological doses, and the appropriate time to manually close the MSIVs.

Sincerely,

Original signed by:

John F. Stang, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-237, 50-249, 50-373,
50-374, 50-254, 50-265

Enclosure: Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 5, 1996

Mr. D. L. Farrar
Manager, Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - GENERIC LETTER 95-07, "PRESSURE LOCKING AND THERMAL BINDING OF SAFETY-RELATED POWER-OPERATED GATE VALVES," ZION STATION, UNITS 1 AND 2 (TAC NOS. M93541 AND M93542) 295, 304 QUAD CITIES STATION, UNITS 1 AND 2 (TAC NOS. M93509 AND M93510), 254, 265 BYRON STATION, UNITS 1 AND 2 (TAC NOS. M93441 AND M93442), AND 454, 455 BRAIDWOOD STATION, UNITS 1 AND 2 (TAC NOS. M93434 AND M93435), 456, 457 DRESDEN STATION, UNITS 2 AND 3 (TAC NOS. M93458 AND M93459), LASALLE 237, 249 COUNTY STATION, UNITS 1 AND 2 (TAC NOS. M93477 AND M93478) 373, 374

Dear Mr. Farrar:

On August 17, 1995, the NRC issued Generic Letter (GL) 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," to request that licensees take actions to ensure that safety-related power-operated gate valves that are susceptible to pressure locking or thermal binding are capable of performing their safety functions. The staff is reviewing and evaluating Commonwealth Edison's response to GL 95-07 dated February 13, 1996. Additional information, as discussed in the enclosure, is requested for the staff to complete its review. This is in addition to the information requested in the staff's letter dated April 2, 1996. This request supersedes the request for LaSalle County Station dated May 20, 1996. We request that you respond within 30 days.

The information requested by this letter is within the scope of the overall burden estimated in Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," which was a maximum of

DF01 1/1

120204

75 hours per response. This request is covered by Office of Management and Budget Clearance Number 3150-0011, which expires July 31, 1997.

Sincerely,

Donna M. Skay

for Clyde Y. Shiraki, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-295, 50-304, 50-254,
50-265, 50-324, 50-454, 50-456,
50-457, 50-237, 50-249, 50-373, 50-374

Enclosure: RAI

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION
ZION STATION, UNITS 1 AND 2, RESPONSE TO GENERIC LETTER 95-07, "PRESSURE LOCKING AND THERMAL BINDING OF SAFETY-RELATED POWER-OPERATED GATE VALVES"

1. Commonwealth Edison's (ComEd's) submittal discusses the potential susceptibility of valves 1(2)SI9011A,B, safety injection (SI) Pump Discharge to reactor coolant system (RCS) Hot Leg, to pressure locking under certain conditions, and states that a thrust calculation was performed which shows that the motor operated valves (MOVs) are capable of opening under pressure locking conditions. Please provide this calculation for the staff's review.

   In addition, ComEd's submittal states that a design change to install a new motor actuator is being reviewed for inclusion in upcoming refueling outages. Please provide specific information and calculations, if applicable, regarding the increase actuator thrust capability as compared to the thrust requirement under pressure locked conditions.

2. Regarding valves 1(2)RC8000A,B, Pressurizer Power Operated Relief Valve Block Valves, ComEd's submittal states that in a steam generator tube rupture scenario, the valves will be opened as quickly as possible after event initiation prior to significant cooldown. Has ComEd determined the postulated RCS pressure at the time the valve would be required to open and completed thrust requirement and actuator capability calculations assuming this pressure? If so, please provide these calculations for the staff's review.

   In addition, ComEd's submittal discusses the potential susceptibility of these valves to thermal binding with respect to low temperature overpressurization protection (LTOP). Commonwealth Edison's submittal states that these valves are not required to perform a safety function prior to implementing LTOP and that the valves are required to open prior to implementing LTOP. This wording is somewhat unclear. Please provide a more detailed explanation of the potential susceptibility of these valves to thermal binding.

ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION
QUAD CITIES STATION, UNITS 1 AND 2, RESPONSE TO GENERIC LETTER 95-07, "PRESSURE LOCKING AND THERMAL BINDING OF SAFETY-RELATED POWER-OPERATED GATE VALVES"

1. Regarding the potential susceptibility of valves 1(2)-2301-3, HPCI Turbine Steam Supply, to thermal binding, Commonwealth Edison's (ComEd's) submittal states that these valves are closed hot after stroke testing or high pressure coolant injection (HPCI) flow testing and remain hot prior to an initiation signal. Does ComEd have test data, such as temperature measurements of the valve body while open and later shut, to verify this assertion? If so, please provide these results for the staff's review.

2. In Attachment 1 to GL 95-07, the staff requested that licensees include consideration of the potential for gate valves to undergo pressure locking or thermal binding during surveillance testing. During workshops on GL 95-07 in each Region, the staff stated that if the closing and subsequent pressure locking or thermal binding of a safety related power operated gate valve during the performance of a test or surveillance would defeat the capability of the safety system or train, the appropriate technical specifications must be followed unless one of the following actions has been taken within the scope of GL 95-07:

   1. Verify that the valve is not susceptible to pressure locking or thermal binding while closed,
   2. Demonstrate that the actuator has sufficient capacity to overcome these phenomena, or
   3. Make appropriate hardware and/or procedural modifications to prevent pressure locking and thermal binding.

   The staff stated that normally open, safety-related power-operated gate valves which are closed for test or surveillance but which must be returned to the open position should be evaluated within the scope of GL 95-07. In Section 5.2.2, Valve Functional Review, ComEd's submittal states that inservice testing (IST) stroke time testing or other surveillances which cycle the valve are not to be included in the review. This appears to be inconsistent with the recommendations of GL 95-07. Please discuss how this specific GL 95-07 concern has been addressed.

3. Through review of operational experience feedback, the staff is aware of instances in which licensees have completed design or procedural modifications to preclude pressure locking or thermal binding which may have had an adverse impact on plant safety due to incomplete or incorrect evaluation of the potential effects of these modifications. Please describe evaluations and training for plant personnel that have been conducted for each design or procedural modification completed to address potential pressure locking or thermal binding concerns.

ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION
BYRON STATION, UNITS 1 AND 2, AND BRAIDWOOD STATION, UNITS 1 AND 2, RESPONSE TO GENERIC LETTER 95-07, "PRESSURE LOCKING AND THERMAL BINDING OF SAFETY-RELATED POWER-OPERATED GATE VALVES"

1. Regarding valves 1(2)RH8716A/B, RHR Crosstie Isolation, Commonwealth Edison's (ComEd's) submittal states that an operability assessment has been completed for these valves which concludes that the valves remain operable and no operability issue exists. Please provide the operability assessment for the staff's review, including any applicable heat transfer, thrust requirement, and actuator capability calculations which may have been performed as part of the operability assessment.

   In addition, the licensee's submittal states that corrective actions will be performed in accordance with the operability assessment. Please explain the corrective actions planned for these valves.

2. Regarding the following valves:

   1(2)RY8000A/B, Pressurizer PORV Isolation
   1(2)SI8801A/B, Charging Pump to RCS Cold Legs Isolation
   1(2)SI8802A/B, SI Pump to RCS Hot Leg Isolation
   1(2)SI8840, RHR to RCS Hot Legs Isolation

   Commonwealth Edison's submittal states that an operability assessment has been completed for these valves, which concludes that the valves remain operable and no operability issue exists. Please provide the operability assessment for the staff's review, including any applicable thrust requirement and actuator capability calculations performed as part of the operability assessment.

3. Through review of operational experience feedback, the staff is aware of instances in which licensees have completed design or procedural modifications to preclude pressure locking or thermal binding which may have had an adverse impact on plant safety due to incomplete or incorrect evaluation of the potential effects of these modifications. Please describe evaluations and training for plant personnel that have been conducted for each design or procedural modification completed to address potential pressure locking or thermal binding concerns.

ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION
DRESDEN STATION, UNITS 2 AND 3, RESPONSE TO GENERIC LETTER 95-07, "PRESSURE LOCKING AND THERMAL BINDING OF SAFETY-RELATED POWER-OPERATED GATE VALVES"

1. Valves 2(3)-2301-36, HPCI Suppression Pool Suction, if flexible-wedge, split-wedge, or double-disk gate valves, may be potentially susceptible to thermally-induced pressure locking caused by heat transfer from the suppression pool during a design basis event. Has the licensee evaluated the potential heat transfer from the suppression pool during a design basis event, and the associated thrust requirement/actuator capability calculations? If so, please provide these evaluations for the staff's review.

2. Valves 2(3)-2301-3, HPCI Turbine Steam Admission, if flexible-wedge, split-wedge, or double-disk gate valves, may be potentially susceptible to thermally-induced pressure locking if they exist in a configuration which may trap steam condensate. In addition, these valves, if flexible-wedge, split-wedge, or solid wedge gate valves, may be potentially susceptible to thermal binding if opened for HPCI testing, shut in a hot condition, allowed to cool, and subsequently required to open at a lower temperature. Please discuss the pressure locking/thermal binding evaluation completed for these valves.

3. In Attachment 1 to GL 95-07, the staff requested that licensees include consideration of the potential for gate valves to undergo pressure locking or thermal binding during surveillance testing. During workshops on GL 95-07 in each Region, the staff stated that if the closing and subsequent pressure locking or thermal binding of a safety related power operated gate valve during the performance of a test or surveillance would defeat the capability of the safety system or train, the appropriate technical specifications must be followed unless one of the following actions has been taken within the scope of GL 95-07:

    1. Verify that the valve is not susceptible to pressure locking or thermal binding while closed,
    2. Demonstrate that the actuator has sufficient capacity to overcome these phenomena, or
    3. Make appropriate hardware and/or procedural modifications to prevent pressure locking and thermal binding.

    The staff stated that normally open, safety-related power-operated gate valves which are closed for test or surveillance but which must be returned to the open position should be evaluated within the scope of GL 95-07. Please discuss if all valves which meet this criterion were included in the review, and the way in which potential pressure locking or thermal binding concerns were addressed.

ENCLOSURE

4. Through review of operational experience feedback, the staff is aware of instances in which licensees have completed design or procedural modifications to preclude pressure locking or thermal binding which may have had an adverse impact on plant safety due to incomplete or incorrect evaluation of the potential effects of these modifications. Please describe evaluations and training for plant personnel that have been conducted for each design or procedural modification completed to address potential pressure locking or thermal binding concerns.

REQUEST FOR ADDITIONAL INFORMATION
LASALLE COUNTY STATION, UNITS 1 AND 2, RESPONSE TO GENERIC LETTER 95-07, "PRESSURE LOCKING AND THERMAL BINDING OF SAFETY-RELATED POWER-OPERATED GATE VALVES"

1. In Attachment 1 to GL 95-07, the staff requested that licensees include consideration of the potential for gate valves to undergo pressure locking or thermal binding during surveillance testing. During workshops on GL 95-07 in each Region, the staff stated that if the closing and subsequent pressure locking or thermal binding of a safety related power operated gate valve during the performance of a test or surveillance would defeat the capability of the safety system or train, the appropriate technical specifications must be followed unless one of the following actions has been taken within the scope of GL 95-07:

    1. Verify that the valve is not susceptible to pressure locking or thermal binding while closed,
    2. Demonstrate that the actuator has sufficient capacity to overcome these phenomena, or
    3. Make appropriate hardware and/or procedural modifications to prevent pressure locking and thermal binding.

    The staff stated that normally open, safety-related power-operated gate valves which are closed for test or surveillance but which must be returned to the open position should be evaluated within the scope of GL 95-07. Please discuss if all valves which meet this criterion were included in the review, and the way in which potential pressure locking or thermal binding concerns were addressed.

2. Through review of operational experience feedback, the staff is aware of instances in which licensees have completed design or procedural modifications to preclude pressure locking or thermal binding which may have had an adverse impact on plant safety due to incomplete or incorrect evaluation of the potential effects of these modifications. Please describe evaluations and training for plant personnel that have been conducted for each design or procedural modification completed to address potential pressure locking or thermal binding concerns.

ENCLOSURE

75 hours per response. This request is covered by Office of Management and Budget Clearance Number 3150-0011, which expires July 31, 1997.

Sincerely,

Original signed by: Donna M. Skay for

Clyde Y. Shiraki, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-295, 50-304, 50-254,
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50-457, 50-237, 50-249, 50-373, 50-374

Enclosure: RAI

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 29, 1996

Mr. D. L. Farrar
Manager, Nuclear Regulatory Services
Commonwealth Edison Company
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1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: REQUEST FOR WITHHOLDING INFORMATION FROM PUBLIC DISCLOSURE

Dear Mr. Farrar:

By letter from Commonwealth Edison Company (ComEd) dated October 2, 1995, and General Electric Company's (GE) affidavit executed by Michael A. Smith dated September 29, 1995, you submitted a proprietary document entitled, "Analysis of the Dresden and Quad Cities Shroud Repair Hardware Seismic Design with Improved Tie Rod and Shroud Weld Crack Equivalent Rotational Stiffness," GE-NE-523-A100-0995, Revision 0, and requested that it be withheld from public disclosure pursuant to 10 CFR 2.790.

GE stated that the information should be considered exempt from mandatory public disclosure for the following reasons:

"(4)a Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies.

(4)b Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

(8) The information identified in paragraph (2), above, [Analysis of the Dresden and Quad Cities Shroud Repair Hardware Seismic Design with Improved Tie Rod and Shroud Weld Crack Equivalent Rotational Stiffness, GE-NE-523-A100-0995, Revision 0] is classified as proprietary because it contains detailed results of analytical models, methods and processes, including computer codes, and it contains the supporting Design Record File (DRF) detailed calculations, results and bases for conclusions. These reports are part of the DRF supporting information to evaluate a hardware design modification (stabilizer for the shroud horizontal welds) intended to be installed in a reactor to resolve the reactor

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DF01 1/1

pressure vessel core shroud weld cracking concern. This detailed level of information usually resides in GENE files, only for audit by customers and the NRC. This information shows in specific detail the processes, codes and methods employed to perform the evaluations summarized in the above identified document. The development and approval of this design modification utilized systems, components, and models and computer codes that were developed at a significant cost to GE, on the order of several hundred thousand dollars."

We have reviewed your submittal and the material in accordance with the requirements of 10 CFR 2.790 and, on the basis of GE's statements, have determined that the submitted information sought to be withheld contains trade secrets or proprietary commercial information.

Therefore, we have determined that the documents entitled, "Analysis of the Dresden and Quad Cities Shroud Repair Hardware Seismic Design with Improved Tie Rod and Shroud Weld Crack Equivalent Rotational Stiffness," GE-NE-523-A100-0995, Revision 0, marked as proprietary will be withheld from public disclosure pursuant to 10 CFR 2.790(b)(5) and Section 103(b) of the Atomic Energy Act of 1954, as amended.

Withholding from public inspection shall not affect the right, if any, of persons properly and directly concerned to inspect the document. If the need arises, we may send copies of this information to our consultants working in this area. We will, of course, insure that the consultants have signed the appropriate agreements for handling proprietary information.

If the basis for withholding this information from public inspection should change in the future such that the information could then be made available for public inspection, you should promptly notify the NRC. You should also understand that the NRC may have cause to review this determination in the future, for example, if the scope of a Freedom of Information Act request includes your information. In all review situations, if the NRC needs additional information from you or makes a determination adverse to the above, you will be notified in advance of any public disclosure.

Sincerely,

/s/

John F. Stang, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-237, 50-249

cc: see next page

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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION (TAC NO. M92914)

Dear Mr. Farrar:

By letter dated June 26, 1995, Commonwealth Edison Company (ComEd) submitted for NRC review, Topical Report NFSR-0111, Revision 0, "BWR Transient Analysis Methods," for Dresden Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Station, Units 1 and 2. The staff is currently reviewing this report and has identified additional information needed to continue its evaluation. The enclosed request for additional information (RAI) requests that ComEd provide the staff with some additional bench-marking information relating to the use of the RETRAN code for reload transient analysis.

Sincerely,

Original signed by:

Donna M. Skay, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-237, 50-249, 50-373,
50-374, 50-254, 50-265

Enclosure: RAI

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REQUEST FOR ADDITIONAL INFORMATION
ON BWR TRANSIENT ANALYSIS METHODS

1. In your topical report dated June 1995 for the LaSalle Reactor Water Level Setpoint Change (RWLSC), you state that core power, steam flow rate, and reactor pressure remain relatively constant as expected over the course of the transient (page 4-5). Provide those results and compare them to the test data, if available.

2. From the several tests/benchmarks presented in the report, pressure discrepancies between the test data and RETRAN02 results could be observed throughout. For example, for the LaSalle Pressure Regulator Setpoint Change (PRSC), test data stabilized 1.5 psi higher than the RETRAN02 results; for the Dual Recirculation Pump Trip (DRPT), a 3 psi difference is observed; for the Main Steam Isolation Valve Closure (MSIVC), the test data stabilized 8 psi lower than the RETRAN02 results; and for the Peach Bottom turbine trip test 2, the reactor dome pressure shows a 3 psi difference.

   Considering that most of the other parameters plotted show superior agreement, discuss why these pressure differences are observed. Where is the pressure parameter measured (both for the test data and in the RETRAN02 model)?

3. In the same report, on page 4-51, you state that "the measured data is clearly in error as the power was measured to level off around 10% after the reactor scram". Discuss/prove that the model results are correct.

4. On page 4-51, you state that the initial rise of the steam flow for the turbine trip with bypass benchmark is not believed to reflect the physical process and represents a temporary error in the flow measurement. Discuss how/why the test data is wrong and describe the expected physical process.

5. On page 6-6, you include the statement "The results show that the RETRAN model would be more conservative." Discuss how you reached this conclusion from the results presented.

ENCLOSURE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DOCKET FILE
50-237/249/254/265
50-295/304/373/374
50-454/455/456/
50-457

April 2, 1996

Mr. D. L. Farrar
Manager, Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR GENERIC LETTER 95-07, "PRESSURE LOCKING AND THERMAL BINDING OF SAFETY-RELATED POWER-OPERATED GATE VALVES," RELATED TO BRAIDWOOD STATION, UNITS 1 AND 2; BYRON STATION, UNITS 1 AND 2; ZION STATION, UNITS 1 AND 2; QUAD CITIES STATION, UNITS 1 AND 2; DRESDEN STATION, UNITS 2 AND 3, AND LASALLE COUNTY STATION, UNITS 1 AND 2 (TAC NOS. M93434, M93435, M93441, M93442, M93458, M93459, M93477, M93478, M93509, M93510, M93541 AND M93542)

Dear Mr. Farrar:

On August 17, 1995, the staff issued Generic Letter (GL) 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," to request that licensees take actions to ensure that safety-related power-operated gate valves that are susceptible to pressure locking or thermal binding are capable of performing their safety function within the current licensing bases of the facility. By letter dated February 13, 1996, Commonwealth Edison Company (ComEd), submitted its 180-day response to GL 95-07 for each of its facilities. Although the staff has not completed its review, it determined that ComEd has developed a methodology to predict the thrust requirement for gate valve actuators to overcome pressure locked conditions and, based on its preliminary understanding, believe it to be a valuable tool. The staff also understands that ComEd has performed testing to validate this methodology and that it is relying on it to justify the design basis capability of certain safety-related power-operated gate valves to perform their safety function within the current licensing bases of the ComEd facilities.

As discussed during a phone call on March 8, 1996, the staff has determined that it requires additional information to complete its review of the program that ComEd developed to address the concerns discussed in GL 95-07. Therefore, submission is requested of the following additional information: (1) the thrust prediction methodology (including the method for predicting actuator output capability), (2) the test procedures (including information specific to each test valve sufficient to perform the pressure locking calculations), (3) the test results (including the method for interpreting diagnostic equipment data), (4) the information regarding the diagnostic equipment used during testing (including calibration methods and diagnostic

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uncertainties), and (5) any limitations or conditions placed on the use of the methodology (i.e., valve size, type, temperature, pressure, etc.)

The Office of Nuclear Regulatory Research is sponsoring tests at the Idaho National Engineering Laboratory to study the effects of pressure locking and thermal binding on selected gate valves. When these test results are made publicly available, the information will be shared with interested licensees.

Upon completing a more thorough review of the ComEd submittal, the staff may request additional information and may also desire to meet with the cognizant members of the ComEd staff regarding its GL 95-07 program.

Sincerely,

Original signed by:

Clyde Y. Shiraki, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-456, STN 50-457,
STN 50-454, STN 50-455, 50-295,
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50-249, 50-373, 50-374

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50-237

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 23, 1996

Mr. D. L. Farrar
Manager, Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNIT 2 - EVALUATION OF CORE SPRAY PIPING INDICATIONS (TAC NO. M93590)

Dear Mr. Farrar:

By letter dated September 12, 1995, Commonwealth Edison Company (ComEd) submitted an evaluation of three indications in the core spray internal piping components identified through in-vessel inspection activities performed during the current refueling outage at Dresden, Unit 2. Additional information was provided by your letter dated September 25, 1995. Based on your evaluation, you concluded that the structural integrity of the core spray internal piping will maintain adequate structural integrity for the next operating cycle without the need to repair the indications.

The inspection of the subject piping was preformed in accordance with the requested actions of NRC IE Bulletin 80-13, "Cracking in Core Spray Spargers," dated May 12, 1980. This Bulletin requires all licensees of operating boiling water reactors to perform a visual inspection of the core spray sparger and the segment of piping between the inlet nozzle and the vessel shroud every refueling outage. Ultrasonic examinations were used to size the length of the flaw indications.

During the visual inspection, crack like indications were visually observed at three components of the core spray downcomer piping. The three flawed components are a lower sparger inlet elbow and an upper and lower sparger inlet thermal sleeve collars. The length of these indications as measured by ultrasonic examinations varied from 2 inches to 5.5 inches. The indications were reported to be very tight and showed characteristics of jagging and branching, which are typical of intergranular stress corrosion cracking.

The staff's Safety Evaluation (SE) concerning the subject flaw indications is enclosed. Based on the SE, the staff concludes that the structural integrity of the subject flawed core spray components will be maintained during the next fuel cycle on the basis that the final flaw sizes at the end of the next fuel cycle will not exceed the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) allowable values. Therefore, Dresden, Unit 2, can be operated safely for the next fuel cycle without repairing the subject flawed core spray piping components. Continued plant operation beyond

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the next fuel cycle should be supported by the results of re-inspection and reevaluation of the subject flaw indications. In addition, to ensure safe plant operation in the long-term, please provide an evaluation to address the plant capabilities in the detection of loose parts during power operation and the program for removing loose parts from the reactor vessel. This evaluation should be provided for staff review prior to restart of the unit from the next scheduled refueling outage.

This completes the NRC staff review of the subject evaluation and closes TAC No. M93590. If you have any questions regarding this issue, please contact me at (301) 415-1345.

Sincerely,

John F. Stang, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-237

Enclosure: Safety Evaluation

cc w/encl: see next page

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UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE FLAW EVALUATION OF THE CORE SPRAY INTERNAL DOWNCOMER PIPING

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNIT 2

DOCKET NO. 50-237

1.0 INTRODUCTION

During the current Dresden, Unit 2, refueling outage (D2R14), crack like indications were visually observed at three components of the core spray internal downcomer piping. The three flawed components are a "B" loop lower sparger inlet elbow, and an upper ("A" loop) and a lower ("B" loop) sparger inlet thermal sleeve collars. All indications were located in the heat affected zones (HAZ) of welds. The flawed piping components were made of type 304 stainless steel and were located inside the vessel annulus between the inside wall of the reactor pressure vessel and the outside wall of the core shroud. The elbow is 6 inches in diameter. Each end of the elbow was welded to the thermal sleeve and the downcomer piping, respectively. The thermal sleeve collar was attached to the outside surface of the core shroud at one end and on the outside surface of the thermal sleeve at the other end. The length of these indications as measured by ultrasonic examination varied from 2 inches to 5.5 inches. The crack indications were reported to be very tight and showed characteristics of jagging and branching. The locations and appearance of these crack indications are typical of intergranular stress corrosion cracking (IGSCC).

By a letter dated September 12, 1995, the licensee submitted flaw evaluation reports of the core spray internal piping for NRC review and approval. The revised flaw evaluation reports were submitted to NRC on September 25, 1995. The revised evaluation reports did not change the conclusions of the previous reports. The results of the licensee's evaluations concluded that sufficient margins exist to operate for one cycle with the identified flaws. The staff's evaluation and conclusion are provided below.

2.0 EVALUATION

Because IGSCC is known to be initiated from the piping inside surface, visual examination can only find flaws that are through-wall. To ensure all flaws, (whether they are through-wall or not) are found and properly sized, the licensee performed ultrasonic examination of each of the flawed core spray components. Because the pipe wall is relatively thin, it is not practical to determine the depth of the flaws and, therefore, only the length of each flaw was ultrasonically determined. Thus, in the licensee's flaw evaluation, each flaw was assumed to be through-wall. The ultrasonic technique used in the examination was developed by General Electric Company (GE) to determine the end points of the detected flaws. The technique was qualified on the mockups of the subject flawed piping components and was independently reviewed by EPRI and the licensee. For the thermal sleeve collars, the UT examination covered 360 degrees of the circumference. The flaw at the upper thermal sleeve collar in loop A was reported to be 2 inches in length. Two flaws were found at the lower thermal sleeve collar in loop B. One of the flaws was not visually observable because it was not connected to the outside surface of the collar. The lengths of the two flaws were reported to be 3 inches and 5.5 inches, respectively. The flaw at the lower sparger inlet elbow in loop B was estimated to be 3.5 inches in length. Due to access limitation, a portion of the elbow circumference (about 4.8 inches) was not ultrasonically examined. However, visual examination did not find any crack indication in this area.

The licensee reported that, based on the fabrication records, the elbow weld was performed using the gas tungsten arc welding (GTAW) process and that the thermal sleeve collar welds were fabricated with the shielded metal arc welding (SMAW) process.

In the crack growth calculation, the licensee used the bounding crack growth rate of $5.0X10^{-5}$ inches/hour. The licensee stated that hydrogen water chemistry (HWC) was implemented at Dresden, Unit 2, since 1983 to mitigate the IGSCC. The licensee also stated that the neutron fluence in the area of the core spray is less than $6.0x10^{+18}$ n/cm^2. Because the neutron fluence is less than the threshold level of $5.0x10^{+20}$ n/cm^2, irradiation assisted stress corrosion cracking (IASCC) is not expected to occur at the subject core spray piping. Based on the consideration discussed above, the staff concludes that the crack growth rate used by the licensee in the crack growth calculation is conservative.

By using the bounding crack growth rate, the licensee calculated the final crack length at the end of the next fuel cycle for a period of 21 months with a 90 percent availability factor (13,608 hours). The final crack length was derived by adding 0.68 inches to each end of the detected flaw.

To develop the loads acting on the thermal sleeve collar flaws, the licensee performed a three dimensional finite element analysis (FEA) by using the ADINA program to model and analyze the core spray thermal sleeve shroud penetration assembly. The results of the FEA (stiffness of the penetration assembly and the load distribution) were used in the PIPSYS program to calculate the loads and stresses in the piping system. The loads used for the elbow flaw evaluation were taken directly from the piping analysis.

The licensee performed the flaw evaluation by using the limit load methodology in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, Appendix C. The ASME Code allows the limit load approach for the welds fabricated by the GTAW process. The loads used in the evaluation were obtained from the piping analysis. The following loads were included in the evaluation: weight, thermal, seismic, operating drag and loss-of-coolant accident (LOCA). The design basis load combinations were evaluated and the worst case of normal/upset and emergency/faulted condition load combinations were used in the evaluations. Additionally, the licensee performed evaluations of cases beyond the design basis faulted condition. The licensee assessed the load design margins and the allowable months of operation for each of these cases. The load design margin is defined as the ratio of the maximum permitted stress to the applied stress. The ratio represents the margin with respect to the applied load above the ASME Code, Section XI, safety factors. The bounding case beyond the design basis was determined to be a simultaneous occurrence of a seismic SSE event and a reactor recirculation line break (RRLB) LOCA. The licensee has determined that the loads generated by the RRLB LOCA event are bounded by the main steam line break (MSLB) LOCA event for this piping.

The results of the licensee's limit load analysis have shown that the bounding final flaw length at the end of the next fuel cycle would not exceed the critical flaw length and that the load margin factor for the bounding design basis condition and the beyond design basis condition is at least 38 and 28, respectively.

The licensee also performed simplified elastic-plastic evaluation for the SMAW welds in accordance with ASME Code, Section XI, Appendix C. The welds at the thermal sleeve collars were fabricated by the SMAW process. In this evaluation, a reduction factor (Z) and the secondary stresses were included in the limit load formulation. At the staff's request, this evaluation was also performed for the elbow weld. In addition, the elbow areas (4.8 inches) that were inaccessible to ultrasonic examination were assumed to be flawed through-wall in this evaluation. The results of the licensee's evaluation showed that the flawed elbow for the condition beyond the design basis represented the bounding case. For the bounding case, the load margin factor was reported to be 1.8. The staff has reviewed the licensee's flaw evaluation and concludes that the licensee's method of evaluation is conservative and complies with the ASME Code requirements and, therefore, the evaluation results are acceptable.

The licensee performed a leak rate calculation for the flawed elbow by using the PICEP program. The thermal sleeve collars are not part of the core spray system pressure boundary and, therefore, are not considered in the core spray system leakage evaluation. The PICEP program was developed by EPRI for leak-before-break applications. The leak rate was calculated for several piping conditions. For the bounding condition of a 64 psig line pressure in the core spray piping with the reactor vessel pressure at a zero psig, the leak rate was calculated to be no more than 1.38 gpm at the end of next fuel cycle and 82.84 gpm at the end of the plant life. The leakage was considered lost in this evaluation as a reactor recirculation suction line break was assumed. The licensee stated that with a concurrent loss of the low pressure coolant injection (LPCI) system, the leakage may impact the peak cladding temperature (PCT). For a core spray leakage of 300 gpm, the licensee's preliminary estimate of the PCT increase is 36 degrees Fahrenheit. Therefore, the licensee concluded that the calculated leakage at the end of the next fuel cycle is well within the design basis margin and its impact on the PCT is insignificant. Since the detected cracks were reported to be very tight, the staff expects the leakage flow resulting from the flawed elbow to be small during the next fuel cycle with no significant impact on the PCT. Therefore, the licensee's conclusion is acceptable for the short term operation of the next fuel cycle.

The licensee performed a safety evaluation of the loose parts which may result from the flawed core spray components. The postulated loose parts consisted of a separated stainless steel elbow and its debris. The safety evaluation considered its potential impact for the fuel bundle flow blockage and consequent fuel damage, fretting wear of the fuel cladding, interference with control rod operation and corrosion or chemical reaction with other reactor materials. The licensee's evaluation concluded that the postulated loose parts would not result in any safety concern in maintaining the proper fuel cooling and the control rod operation. Although extensive IGSCC may lead to the separation of pieces of various sizes from the flawed components, in the short term, the staff does not anticipate any loose parts to occur; especially the separation of the elbow. However, to ensure safe plant operation in the longer term, the staff recommends that the licensee submit an evaluation prior to the end of the next refueling outage to address the plant capabilities in the detection of the loose parts during operation and the program for removing the loose parts from the reactor pressure vessel.

3.0 CONCLUSION

Based on the staff's review of the licensee's flaw evaluations, the staff concludes that the structural integrity of the subject flawed core spray components will be maintained during the next fuel cycle on the basis that the final flaw sizes at the end of the next fuel cycle will not exceed the ASME Code allowable values. Therefore, Dresden, Unit 2, can be safely operated for the next fuel cycle without repairing the subject flawed core spray components. However, continued plant operation beyond the next fuel cycle will depend on the satisfactory evaluation of the re-inspection results or by implementing acceptable repairs during the next refueling outage.

Principle Contributor: Bill Koo

Date: February 23, 1996

4.0 REFERENCES

Letter, Peter L. Piet, Commonwealth Edison Company, to U.S. Nuclear Regulatory Commission, September 12, 1995.

Letter, Peter L. Piet, Commonwealth Edison Company, to U.S. Nuclear Regulatory Commission, September 25, 1995.

the next fuel cycle should be supported by the results of re-inspection and reevaluation of the subject flaw indications. In addition, to ensure safe plant operation in the long-term, please provide an evaluation to address the plant capabilities in the detection of loose parts during power operation and the program for removing loose parts from the reactor vessel. This evaluation should be provided for staff review prior to restart of the unit from the next scheduled refueling outage.

This completes the NRC staff review of the subject evaluation and closes TAC No. M93590. If you have any questions regarding this issue, please contact me at (301) 415-1345.

Sincerely,

/s/

John F. Stang, Senior Project Manager
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Docket No. 50-237

Enclosure: Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 6, 1995

Mr. D. L. Farrar
Manager, Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: SAFETY EVALUATION REGARDING CORE SHROUD REPAIR - DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3 (TAC NOS. M91301, M91302 AND M93584)

Dear Mr. Farrar:

On July 25, 1994, the staff issued Generic Letter (GL) 94-03 concerning core shroud cracking in boiling water reactors (BWRs). By letter dated March 30, 1995, Commonwealth Edison Company (ComEd, the licensee) responded to the generic letter and submitted the inspection plan for the Dresden Nuclear Power Station, Unit 2, core shroud (GL 94-03 Item 2.(a)). By letter dated May 24, 1995, ComEd submitted the design documents for the repair of the Dresden, Units 2 and 3, core shrouds (GL 94-03 Item 2.(b)).

As a result of the review of the licensee's repair design submittal, the staff requested additional information (RAI) on July 26, 1995, and held telephone discussions with the licensee on August 31, 1995. The licensee provided its response to the staff's RAI in separate submittals on August 14, September 5, September 25 and October 2, 1995. This response also included the ComEd 10 CFR 50.59 safety evaluation of the core shroud. The licensee's submittal dated August 28, 1995, provided the final results of the Dresden, Unit 2, core shroud examination.

The proposed core shroud repair has been designed as an alternative to the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI. Pursuant to 10 CFR 50.55a(a)(3)(i), use of an alternative to the ASME Code requires review and approval of this repair by the NRC staff.

The staff has reviewed the structural aspects of the proposed repair provided in the licensee's submittals of May 24, July 26, August 14, September 5, September 25 and October 2, 1995. Our evaluation is provided in the enclosed safety evaluation. Based on a review of the shroud modification hardware from structural, systems, materials, and fabrication considerations, the staff concludes that the proposed modifications of the Dresden, Units 2 and 3, core shroud are acceptable and will not result in any increased risk to the public health and safety. In accordance with 10 CFR 50.59, ComEd determined that no unreviewed safety question will result and no technical specification revision will be involved. The staff agrees with this determination and concludes that no license amendment, pursuant to 10 CFR 50.90, is necessary.

This completes our action with respect to the above TACs.

Sincerely,

Original Signed By:

John F. Stang, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-237, 50-249

Enclosure: Safety Evaluation

cc w/encl: see next page

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UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE PROPOSED REPAIR FOR THE CORE SHROUD

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

DOCKET NOS. 50-237 AND 50-249

1.0 BACKGROUND

In Boiling Water Reactors (BWR) the core shroud is a stainless steel cylinder within the reactor pressure vessel (RPV) that provides lateral support to the fuel assembly. The core shroud also serves to partition feedwater in the reactor vessel's downcomer annulus region from cooling water flowing through the reactor core. The RPV, core shroud and other RPV internals are designed to accomplish three basic safety functions:

- provide a refloodable coolant volume for the reactor core to assure adequate core cooling in the event of a nuclear process barrier breach;
- limit deflections and deformation of internal safety-related RPV components to assure that control rods and emergency core cooling systems (ECCS) can perform their safety functions during anticipated operational transients and/or design basis accidents;
- assure that the safety functions of the core internals are satisfied with respect to safe shutdown of the reactor and proper removal of decay heat.

In 1991, cracking of the core shroud was visually observed in a foreign BWR. The crack in this BWR was located in the heat-affected zone of a circumferential weld in the mid-core shroud shell. The General Electric Company (GE) reported the cracking found in the foreign reactor in Rapid Information Communication Services Information Letter (RICSIL) 054. GE identified the cracking mechanism as intergranular stress corrosion cracking (IGSCC).

A number of domestic BWR licensees have recently performed visual examinations of their core shrouds in accordance with the recommendations in GE RICSIL 054 or in GE Services Information Letter (SIL) 572, which was issued in late 1993 to incorporate domestic experience. The combined industry experience from plants which have performed inspections to date, indicates that both axial

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and circumferential cracking can occur in the core shrouds of GE designed BWRs, and that extensive cracking can occur in circumferential welds located both in the upper and lower portions of BWR core shrouds. The cracking reported in the Brunswick, Unit 1, core shroud was particularly significant since it was the first time that extensive 360 degree core shroud cracking had been reported by a licensee in a domestic BWR. The 360 degree core shroud crack at Brunswick, Unit 1, was located at weld H3 which joins the top guide support ring to the mid-core shroud shell. Information Notice 93-79 was issued by the NRC on September 30, 1993, in response to the observed cracking at Brunswick Unit 1.

The cracks reported by the Commonwealth Edison Company (ComEd) in the Dresden, Unit 3, and Quad Cities, Unit 1, core shrouds were of major importance, since they signified the first reports of 360 degree cracking located in lower portions of BWR core shrouds. These 360 degree cracks are located at core shroud weld H5, which joins the core plate support ring to the middle core shroud shell in both the Dresden and Quad Cities Units. Information Notice 94-42 and its Supplement were issued by the NRC on June 7 and July 19, 1994, to alert other licensees of the core shroud cracking discovered at Dresden, Unit 3, and Quad Cities, Unit 1.

On July 25, 1994, the NRC issued Generic Letter (GL) 94-03 (Reference 1) to all BWR licensees (with the exception of Big Rock Point, which does not have a core shroud) to address the potential for cracking in their core shrouds. GL 94-03 requested BWR licensees to take the following actions with respect to their core shrouds:

- inspect the core shrouds no later than the next scheduled refueling outage;
- perform a safety analysis supporting continued operation of the facility until the inspections are conducted;
- develop an inspection plan which addresses inspections of all core shroud welds and which delineates the examination methods to be used for the inspections of the core shroud, taking into consideration the best industry technology and inspection experience to date on the subject;
- develop plans for evaluation and/or repair of the core shroud and work closely with the Boiling Water Reactor Owners' Group (BWROG) on coordination of inspections, evaluations, and repair options for all BWR internals susceptible to IGSCC.

By letters dated May 24 (Reference 3), July 26 (Reference 4), August 14 (Reference 5), September 5 (Reference 6), September 25 (Reference 7) and October 2, 1995 (Reference 8), ComEd responded to GL 94-03 by submitting the details of the planned repair of the Dresden, Units 2 and 3, core shrouds. Part of the licensee's response included ComEd's plans for inspection of the Dresden, Unit 2, core shroud during the upcoming refueling outage and plans for a repair that involves a permanent modification. ComEd advised the staff that the modification will encompass the entire set of circumferential welds in the core shroud and will involve the installation of four (4) restraint assemblies in the annulus region around the core shroud.

2.0 EVALUATION

2.1 Scope of the Modification Design

The scope of this safety evaluation (SE) focuses on the circumferential welds in the core shroud, since the only significant cracking of BWR core shrouds has been associated with these welds. The staff is currently not aware of any extensive cracking of vertical seam welds in BWR core shrouds. As stated in Section 2.5.2, ComEd also inspected the vertical welds and determined that cracking in these welds has been limited to relatively small lengths. However, based on industry experience, vertical weld cracks less than three (3) inches (with one exception, where the crack length was 15 inches) have been observed elsewhere.

The Dresden core shroud repair has been designed to restrain the core shroud head, the top guide support ring, the core and core plate support ring, and to prevent upward displacement of the core shroud during postulated accident conditions. The modification has been designed as an alternative to the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) pursuant to 10 CFR 50.55a(a)(3)(i). It is designed to structurally replace the circumferential welds from the H1 weld at the top of the core shroud to the H7 weld at the bottom of the core shroud. The Dresden core shroud repair design, therefore, provides structural integrity for, and takes the place of, all circumferential welds which are subject to cracking in the Dresden core shrouds. ComEd has also stated that the repair is designed for 40 years, including 30 effective full power years. This indicates that the design of the repair accounted for the remaining life of the plant plus possible life extension beyond the current operating license.

Details of the modification are contained in a number of GE proprietary reports which were reviewed by the staff. These are contained in References 3 through 8.

2.2 Core Shroud Repair Modification Description

The design of the Dresden, Units 2 and 3, core shroud repair consists of four (4) tie rod stabilizer assemblies, which are installed 90 degrees apart in the core shroud/reactor vessel annulus, between attachment points at the top of the core shroud flange and toggle support assemblies attached to the core shroud support plate. Each tie rod stabilizer assembly consists of upper, middle and lower spring assemblies connected by a solid rod. The rod provides the vertical load transfer from the core shroud head flange to the core shroud support plate attachment and supports the spring assemblies. The upper spring assembly provides lateral load support at the top guide elevation from the core shroud to the RPV. The lower spring assembly provides lateral support from the core shroud at the core plate support ring elevation to the RPV. The middle spring assembly provides lateral support for the mid sections of the core shroud and increases the natural frequency of the tie rod stabilizer to reduce flow induced vibration. Each cylindrical section of the core shroud between welds H1 through H7 is prevented from unacceptable lateral motion by these tie rod stabilizer assemblies.

The upper spring assemblies of the tie rod stabilizer assemblies are attached to the core shroud head flange by means of brackets which are installed into slots machined in the flange. The lower end of the tie rod stabilizer assemblies are attached to pins in toggle assemblies which are bolted into holes cut into the core shroud support plate. Hook devices on the lower spring assemblies allow attachment to the toggle assemblies. The tie rod stabilizer assemblies provide vertical restraint to the core shroud. The springs limit the lateral displacements of the core shroud during horizontal dynamic loading in the postulated event of a 360 degree through-wall failure of one or more of the circumferential welds, so as to ensure control rod insertion. Together, the tie rod stabilizer assemblies and the lateral restraints resist both vertical and lateral loads resulting from normal operation and design accident loads, including seismic loads and postulated pipe ruptures.

The tie rod stabilizer assemblies are installed with a small vertical preload such that the core shroud is in compression during cold shutdown conditions. The coefficients of thermal expansion of the components of the tie rod stabilizer are smaller than those of the core shroud such that the compressive preload on the core shroud increases as the reactor reaches operating conditions. The combined spring constant of the tie rod stabilizer assemblies and the core shroud together, was designed to provide a total vertical preload at operating conditions which will assure no separation of any or all failed circumferential welds from H1 through H7 during normal plant operation. Vertical separation for any and all welds is precluded except for the postulated design event consisting of a main steam line break loss-of-coolant accident (LOCA) combined with a design basis earthquake, since excessive preload would be required to prevent any separation for this event. Similarly, the upper, middle and lower spring assemblies are installed with a small preload during cold shutdown. During normal operation, the lateral expansion of the core shroud and the spring assemblies due to thermal growth is greater than that of the RPV, providing additional preload and support for the core shroud. This preload will restrict the lateral core shroud displacements during postulated accident conditions within acceptable limits and assure prompt rod insertion during these conditions.

2.3 Structural Evaluation

2.3.1 Core Shroud and Tie Rod Stabilizer Assemblies

The repair of the core shroud using the tie rod stabilizer assemblies have been designed to the structural criteria specified in the Dresden Updated

Final Safety Analysis Report (UFSAR) (Reference 9). The seismic analyses were performed in accordance with the methods described in the UFSAR. All of the loads and load combinations specified in the UFSAR which are relevant to the core shroud were included in the design. The tie rod stabilizer assemblies were designed using the ASME Code Section III, 1989 Edition, Subsections NB and NG as a guide (Reference 10). The original ASME Code Section III (1965 Edition with June 30, 1966, Addenda thru Summer 1965) for the design and construction of the RPV did not have design requirements for core support structures. The additional loads placed on the RPV by the stabilizer assemblies have been evaluated to the original design Code.

ComEd evaluated all load combinations required by the UFSAR for normal, upset, emergency, and faulted conditions which include: normal (dead weight (DW) plus normal operating temperature), thermal upset, Operating Basis Earthquake (OBE), Design Basis Earthquake (DBE), Main Steamline Break (MSLB) LOCA, and Recirculation Line Break (RLB) LOCA loads. All internal loads including those due to the two faulted load combinations of DBE plus LOCA were combined by absolute summation. A three-dimensional finite element analysis model was developed for the stress analysis of the core shroud and the tie rod stabilizer assemblies (References 11, 12 and 13). The analysis was performed using the commercial finite element program ANSYS (Reference 14). The use of ANSYS for modelling of the core shroud and the tie rod stabilizer assemblies is acceptable to the staff. ComEd evaluated the dynamic nature of the DBE, RLB and MSLB LOCA loads on the repaired core shroud structure. The RLB LOCA lateral loading fluctuates with time, but the initial acoustic loading has an input frequency much greater than the core shroud frequency content such that there is very little response due to the initial acoustic loading. ComEd determined that the portion of the RLB loading following the acoustic portion is relatively constant which would result in a static load with no amplification, and that the RLB loads were bounded by the MSLB loads for the design of the stabilizer.

The limiting upset loading condition event which ComEd evaluated is the cold feedwater transient which is classified as an upset loading condition. During this transient, due to injection of cold feedwater into the core shroud annulus, a maximum temperature difference of 133 degrees Fahrenheit between the hot core shroud and the cooler tie rod stabilizer assembly components could exist. This would cause an increase in the tensile load on the stabilizer and an increase in the compressive load on the core shroud. ComEd evaluated this condition and determined that the stresses in the stabilizer and in the core shroud for this condition would be both less than the ASME Code upset allowable stress and less than the material yield stress, thus preventing permanent deformation, which is acceptable. ComEd also determined that this event is the only case which produces any fatigue in need of consideration. For this event, the maximum calculated fatigue usage was found to be insignificant compared to the allowable usage and is, therefore, acceptable.

ComEd has also investigated the effects of radiation on the repair design. Specifically, ComEd determined that the fast flux levels on the stabilizer are low compared to levels which could degrade material properties. Further, the service temperature for this application has no significant effect on the degradation of the repair materials.

The NRC staff has reviewed the methodology and results of the stress analysis of the core shroud and tie rod stabilizer assembly and has determined it meets the appropriate criteria to assure core shroud structural integrity and, therefore, is acceptable.

2.3.2 Evaluation of Postulated Critical Weld Failures

ComEd evaluated an enveloping combination of postulated cracked/uncracked welds to define the worst case for the core plate and top guide displacements to ensure control rod insertion and safe shutdown during the assumed normal, upset, emergency and faulted conditions required by the UFSAR. Each postulated through-wall cracked weld was modelled as a hinge or roller to determine the limiting displacement. In References 15 and 16, ComEd provided the maximum allowable transient and permanent displacements of the core plate and top guide. Justification for these allowable displacements is provided in Reference 26. The staff agrees that these maximum displacements are reasonable and acceptable. The predicted worst case lateral transient deflection of the core plate support ring during a DBE is less than the allowable limit of 1.12 inches. The worst lateral transient displacement of the top guide support ring during an DBE is also substantially less than the allowable limit of 3.6 inches.

The limiting loads in the tie rod stabilizer assemblies and the limiting loads in the upper, middle and lower springs occur for different assumed core shroud crack combinations (Reference 15). The limiting loads in the tie rod stabilizer assemblies occur under the 1940 El Centro DBE plus operating pressure, assuming a through-wall crack in weld H4 when it behaves as a hinge. The limiting loads in the radial direction on the upper and lower springs occur under the Housner DBE plus operating pressure where it is assumed that all horizontal welds in the core shroud are cracked and represented as hinges. The limiting load in the radial direction on the middle spring occurs under the Housner DBE plus MSLB LOCA where it was assumed that all horizontal welds in the core shroud are cracked and represented as hinges except for H1, which was represented as a roller. The middle spring is designed to prevent radial deflections of the core shroud from exceeding acceptable limits. The upper and lower springs are similarly designed to prevent the radial deflection of the top guide support ring and the core plate support ring from exceeding acceptable limits.

The tie rod stabilizer assembly preload prevents the vertical separation of the core shroud at all potential crack locations during normal operation. The critical cracked weld locations are for H2 and H3 since the failure of these welds has a significant effect on the vertical stiffness of the core shroud due to the greater deflections in the top guide support ring when vertical loads are applied. ComEd also included the effect of a postulated failure of the H5 and H6 welds on the vertical core shroud stiffness. The most severe consequences are determined to occur if these welds are postulated to be initially intact, but fail subsequently in operation. For this scenario, ComEd's calculations indicate that there is sufficient preload to prevent weld separation due to the change in rigidity of the core shroud structure. ComEd determined that the tie rod stabilizer assembly cold preload could be reduced to zero due to the application of the core shroud head weight when it is installed if the core shroud stiffness is reduced the maximum amount. However, since the mechanical cold preload is only a small part of the total hot operating preload, there will be no separation at any welds during normal operation. The staff has reviewed ComEd's evaluation and finds it reasonable and acceptable.

In Reference 5, ComEd reported that the maximum expected vertical separation of the H7 weld at the 180 degree azimuth would be 0.452 inch for the postulated DBE plus dead weight plus operating pressure and temperature load combination. This displacement is momentary since the tie rod stabilizer assemblies and the weight of the core shroud and the internals will close the gap once the event is over. This value was based on the maximum tie rod stabilizer assembly load determined from the 1940 El Centro DBE plus normal pressure analysis considering weld H4 cracked as a hinge (References 15 and 16). ComEd also stated that the core spray piping does not provide significant restraint to the core shroud vertical movement during this load combination, and that this piping will remain operable for this postulated single occurrence. The staff finds these results reasonable and acceptable.

2.3.3 Seismic Analysis

A two-dimensional linear elastic dynamic analysis (References 15 and 16) of coupled structural stick models of the Turbine Building, the Reactor Building, the RPV and the reactor internals subjected to horizontal seismic excitation was performed consistent with the original design methods and the original analysis in the UFSAR. Both East-West and North-South seismic models were analyzed. With the exception of the nuclear core and the core shroud (including the repair hardware), these models were identical to the original seismic models. The seismic models incorporated the tie rod stabilizer assemblies and the core shroud with postulated 360 degree thru-wall cracks. The tie rod stabilizer assemblies were modeled as an equivalent rotational spring and incorporated into the stick model, and these were assumed to resist the horizontal seismic loading acting on the core shroud. However, due to the postulated cracked welds, the structural behavior of the core shroud is non-linear, with different mass and stiffness characteristics causing the dynamic properties of the core support shroud and the tie rod stabilizer assemblies to vary, depending on the particular load combination and the postulated cracked weld configuration. To permit the application of linear elastic analysis, the core shroud was represented by a number of stick models, in which the critical cracked welds were represented by hinges or rollers. For the emergency loading condition of DBE plus operating pressure, the maximum load in the highest loaded tie rod stabilizer assembly was determined if the core shroud was postulated to be cracked at the H4 weld, and this weld was represented as a hinge. For the faulted loading condition of DBE and MSLB LOCA, the maximum load in the highest loaded tie rod stabilizer assembly was determined if the core shroud was postulated to crack at the H3 weld, and the H3 weld was assumed to be represented by a roller. Seismic analyses were performed considering these loading conditions and core shroud models as bounding cases. These analyses were performed using the GE proprietary computer program SAP4G07 (Reference 17) that has been accepted for this application.

The seismic analysis for the OBE and DBE is based on time history ground motion input. Two horizontal earthquake time histories were applied to the structural model at the mat foundation and used to generate DBE seismic design loads for the core shroud repair: (1) a synthetic time history whose response spectrum envelopes the Housner seismic response spectrum, and (2) the N-S component of the 1940 El Centro earthquake time history. Both time histories have a normalized peak ground acceleration of 0.20g. These time histories were used for consistency with the original design as stated in the UFSAR. The USFAR material damping ratios were used in the analysis (corresponding to percent of critical damping) and are the same for both OBE and DBE conditions. The seismic analyses were performed for the DBE condition only, and the OBE seismic loads were taken as half of the DBE loads.

In order to account for uncertainties in the seismic input and modelling of the core shroud repair, ComEd included some conservatism in the time history input ground motion for the artificial Housner and El Centro earthquakes. The response spectra from both of these time histories envelope the smoothed Housner UFSAR spectra used as a target. ComEd stated that the duration of the synthetic Housner time history was increased to 40 seconds which increases the energy content of the input ground motion.

Forces and moments due to vertical seismic loading were calculated by using the vertical zero period acceleration (ZPA) equal to 0.13g (2/3 of 0.20g) for DBE as the multiplier of the dead weight which is also consistent with the original design methods. The seismic design loads which were used for the design and analysis of the repair hardware was bounded by the higher of the Housner or 1940 El Centro responses. The peak horizontal and vertical seismic loads were combined by absolute summation with other loads in the core shroud and the repair hardware analyses.

During the review of the seismic analyses for the Dresden, Units 2 and 3, shroud repair hardware design, a discrepancy was discovered in the original 1968 GE seismic report which was used to reconstruct the primary structure seismic models utilized in those analyses. In the 1968 report, the mass corresponding to the top guide node was incorrectly listed as 1.73E3 slugs as opposed to the correct value of 17.33E3 slugs. Consequently, reanalysis was performed to reconfirm the seismic design adequacy of the existing shroud repair hardware design as well as other RPV and internals components (e.g., fuel, guide tubes, CRDs, etc.) and the vessel major supports (i.e., the RPV skirt and stabilizer and the star-truss).

The licensee used a new methodology for representing the shroud weld cracks in the revised seismic analysis. The "pinned" and "roller" weld crack conditions utilized in the initial shroud repair design were replaced with a pinned node in conjunction with a rotational spring at each weld crack location in the shroud. The representation also results in significant reductions in the shroud repair hardware design loads for the same seismic excitation. Thus, significantly-higher seismic design margins can be demonstrated for the existing hardware design.

The revised seismic analysis for the RPV internals with the core shroud repair hardware installed is provided in report GENE-523-A100-0995 (Reference 8, Appendix A). This report, which incorporates the revised hydrodynamic mass, provides the analysis approach, methodology and results regarding the revised seismic analysis of the Dresden and Quad Cities plants with the core shroud repair hardware installed. Based on its review of these new seismic analyses, the staff finds that the loads previously used for the design of the core shroud repair are larger and, thus, bound the new results and are, therefore, acceptable. While all of the results for the core shroud repair hardware were bounded by the original analyses, the loads on some of the internals increased. The effect of these load increases were evaluated and found to be within the existing design margin. A comparison of the nodal frequencies and nodal participation factors obtained from the earlier analyses with the incorrect mass and the revised seismic analysis shows that the effect of the nodal mass discrepancy is minimal with respect to the overall seismic response.

The staff has reviewed the methodology and results of the seismic analysis of the core shroud and the repair hardware and has found them to be plausible and in accordance with current seismic analysis practice and, therefore, acceptable.

2.3.4 Evaluation of RPV Components

ComEd performed an evaluation (Reference 18) of the core shroud support plate stresses in the vicinity of the tie rod stabilizer bolt attachments with the H8 weld both cracked and uncracked, using a detailed finite element model and the ANSYS code. ComEd also computed the effect of the additional loads from the core shroud repair on the original RPV design, including the core shroud support legs (References 20 and 21). The stresses were evaluated for the combined loading of weight, pressure differential and the tie rod stabilizer loading, resulting from the specified operating, emergency and faulted conditions. The stresses were shown to be within the ASME Code allowable stresses. A fatigue analysis was also performed which showed that the usage factor resulting from the upset thermal condition is minimal. The staff has reviewed these results and finds them reasonable and acceptable.

ComEd also addressed the core plate preload clamping force adequacy against lateral sliding relative to the core plate support ring under horizontal DBE seismic forces and resultant vertical loading due to dead weight, buoyancy, vertical DBE and the pressure difference induced by MSLB LOCA (Reference 18). The results indicate that the clamping force is adequate to resist sliding, and that no wedges are needed to prevent sliding. The staff has reviewed these results and finds them reasonable and acceptable.

2.3.5 Potential for Flow-Induced Vibration

ComEd evaluated the potential for flow-induced vibration by calculating the lowest natural frequency of the tie rod stabilizer and the highest vortex shedding frequency due to the water flow in the core shroud annulus. ComEd found that the lowest natural frequency of the tie rod stabilizer assemblies is 37.8 Hertz while the maximum vortex shedding frequency is 4.6 Hertz. Therefore, ComEd determined that there would be essentially no resulting flow-induced vibration fatigue of any of the tie rod stabilizer assembly components. The staff finds these results reasonable and acceptable.

2.3.6 Loose Parts Considerations

ComEd stated that all components of the tie rod stabilizer assemblies will be locked in place with mechanical devices and that loose pieces can not occur without the failure of a locking device. Further, ComEd determined that if a tie rod stabilizer assembly were to fail during normal operation, the leakage through any through-wall cracks would increase, but would not be detectable. If the failed tie rod stabilizer assembly part came completely loose, it could fall onto the core shroud support plate or be swept into the recirculation pump suction line. ComEd stated that the consequences of such a loose part would be consistent with other postulated loose parts. If ComEd's tie rod stabilizer assembly inspection results, following the first fuel cycle of operation, indicate that further measures are necessary to assure that the tie rod stabilizer assemblies (or parts thereof) will not become loose or detached during plant operation, ComEd will be required to augment the inservice inspection plan to address these additional measures.

ComEd stated that full-scale mock ups, which actually represent the plant core shroud and vessel configuration, have been used to qualify and train personnel for the stabilizer assembly installation task. To install the stabilizer, it is necessary to cut and hone holes in the core shroud support plate and to cut notches in the core shroud head flange using the electric discharge machining (EDM) process. The EDM equipment collects about 95 percent of the swarf generated during the machining. ComEd evaluated the impact which the remaining metal particles/filings would have on reactor operation and determined that the suspended particles will be carried away to the reactor water cleanup (RWCU) system where they will be removed and will not increase any short- or long-term degradation of the CRD or recirculation pump wear.

2.3.7 ComEd's 10 CFR 50.59 SE of Core Shroud Repair

In Reference 24, ComEd provided its 10 CFR 50.59 SE of the core shroud repair. In accordance with 10 CFR 50.59, ComEd determined that no unreviewed safety question will result and no technical specification revision will be involved as a result of the implementation of the core shroud repair. The staff agrees with this determination, and concludes that no license amendment, pursuant to 10 CFR 50.90, is necessary.

2.3.8 Conclusion

ComEd has demonstrated that the maximum stresses in the core shroud and the tie rod stabilizer assemblies resulting from operating, upset thermal and emergency and faulted accident conditions meet the corresponding ASME Code-allowable stresses. The staff has reviewed the referenced documents and has determined that the results are reasonable and in general agreement with design and analysis practices employed in support of other core shroud repairs reviewed by the staff. Based on the foregoing discussion, the staff, therefore, concludes that the proposed core shroud repair modification is acceptable from a structural standpoint.

2.4 Systems Evaluation

The Systems evaluation relates to the system-induced leakage, shroud weld crack leakage, downcomer flow characteristics, lateral and vertical displacements. In these areas, the analytical results have been reviewed against the results of the revised consequence assessment without the shroud repair dated December 14, 1994 (Reference 22).

2.4.1 Tie Rod Stabilizer Assembly System Induced Leakage

The installation of the tie rod stabilizer assemblies requires the machining of eight holes through the core shroud support plate using the EDM process. The licensee estimates that a small amount of core flow leakage will occur through the clearance slots. The total calculated leakage from the installation of the tie-rod stabilizer assemblies was estimated to be 0.12 percent of core flow (325 gpm) at 100 percent rated power and 100 percent rated core flow (Reference 23). The staff does not consider this leakage rate to be significant with regards to total core flow and, therefore, it is acceptable.

The installation of the tie-rod stabilizer assemblies also requires the machining of eight pockets into the shroud head flange in order to install the long upper supports. The pockets are machined into the core shroud head flange leaving 0.5 inches of core shroud head flange material at the back of the pocket. The shroud head flange is located above the H1 weld which is the uppermost weld on the shroud and is above the top guide. At this location, core flow is considered to be two-phase flow. Leakage at this location does not bypass the core and, therefore, is acceptable.

At Dresden, the ECCS consists of the single-train high pressure coolant injection (HPCI) system, the automatic depressurization system (ADS), the two-train core spray (CS) system, and the two-train low pressure coolant injection (LPCI) system. The staff notes that the leakage from the shroud support plate and the shroud head flange to the downcomer annulus does not affect the performance of the above systems. Therefore, the ECCS performance is not affected by the physical installation of the tie-rod system.

2.4.2 Shroud Weld Crack Leakage

The tie-rods are installed with a cold preload to ensure that no vertical separation of any or all cracked horizontal welds will occur during normal operations. Vertical separation, if sufficiently large, could compromise fuel geometry and control rod insertion. For Dresden, a maximum vertical separation of 15 inches is required for the top guide to clear the top of the fuel channels. Without the repair, the licensee estimated that there would be no vertical separation during normal operation at the H3 weld location assuming 360 degree through-wall weld failure (Reference 22). With the repair, the licensee stated that the preload on the tie-rods will not allow vertical separation of failed welds during normal operations. The staff notes that, with or without the repair, the estimated vertical separation during normal operations will not affect the fuel geometry and, therefore, control rod insertion is not precluded. However, a small leakage path could exist due to existing through-wall shroud weld cracks. The licensee conservatively modeled the crack to provide a 0.001 inch leakage path per weld, H1 through H8. The licensee estimated that the total leakage from all welds, H1 through H8, having postulated 360 degree through-wall cracks was approximately 140 gpm (0.04 percent of core flow) at 100 percent rated power and 100 percent rated core flow (Reference 23). Although shroud crack leakage is unlikely due to the preload on the tie-rod, the licensee concluded that there are no consequences associated with the repair installed based on these small leakages during normal operations. The staff acknowledges that the total leakage is insignificant and will not affect the performance of the ECCS.

2.4.3 Downcomer Flow Characteristics

The licensee analyzed the available flow area in the downcomer with the four tie-rod assemblies installed. The licensee stated that the size of the tie-rod assemblies is small compared to the size of the jet pump assemblies and thus, the tie-rod assemblies are not expected to significantly affect the flow characteristics in the downcomer. However, since the downcomer annulus is smaller at the top of the shroud with other existing obstructions such as the core spray lines, the licensee evaluated the flow blockage area at one elevation of the upper core shroud restraint of the tie-rod stabilizer assembly. This realistic calculation demonstrated that the installation of the tie-rod stabilizer assemblies will decrease the available downcomer flow area by approximately 2 percent at the top of the core shroud (Reference 24). The staff requested the licensee to perform a more conservative calculation using the plan view of the upper core shroud restraint assembly and existing downcomer hardware.

The licensee's second analysis demonstrated that the installation of the tie-rod stabilizer assemblies will decrease the available downcomer flow area by approximately 10.6 percent (Reference 25). The staff reviewed both downcomer flow calculations for the upper annulus area which accounted for the core spray piping, miscellaneous bolts, lugs, and brackets, and the upper support and spring of the tie-rod assemblies. The staff notes that, consistent with design requirements, the upper core shroud restraint assembly is much larger than any other previous GE repair design (except Quad Cities, Units 1 and 2) and that the 10.6 percent decrease in downcomer flow area is comparable with repair designs reviewed by the staff for other facilities. Based on the licensee's analyses, the staff concluded that the installation of the tie-rod assemblies will not have a significant impact on the downcomer flow characteristics. Additionally, the licensee provided the corresponding pressure drop to the decrease in downcomer flow area. The licensee estimated that the loop pressure drop due to the installation of tie-rod assemblies is negligible. Based on this information and information from other reviews of similar core shroud repairs, the staff concluded that the impact on the loop pressure drop is insignificant. Therefore, the staff agrees with the licensee that the installation of the tie-rod assemblies should not affect the recirculation flow of the reactor.

2.4.4 Potential Lateral Displacement of the Shroud

The licensee also evaluated the maximum lateral displacement of the shroud at the core support plate and top guide under normal operations and load combinations such as DBE, MSLB, and RLB. Lateral displacement of the shroud could damage core spray lines and could produce an opening in the shroud, inducing shroud bypass leakage and complicating recovery. Maximum permanent displacements of the shroud are limited by the restoring force of the lateral springs and was calculated to be minimal for normal and worst case accident scenarios. This lateral displacement is significantly less than the 2-inch thickness of the shroud, and accordingly, the separated portions of the shroud would remain overlapped during worst case conditions.

Additionally, a permanent lateral displacement of the top guide or core plate to the actual magnitude shown in the submittal will not significantly increase the scram time as demonstrated in Reference 26. Therefore, the staff has concluded that the maximum lateral displacement of the core shroud would not result in significant leakage from the core to the downcomer region following an accident scenario and the ability to reflood the core to 2/3 core height would not be precluded.

2.4.5 Potential Vertical Separation of the Shroud

The licensee evaluated the maximum vertical displacement of the shroud assuming 360 degree through-wall cracks at any weld above or below the core support plate during a MSLB and a MSLB plus DBE. These postulated events would result in a large upward load on the shroud which could impact the ability of the control rods to insert and the ability of the core spray system to perform its safety function. As stated above, a maximum vertical separation of 15 inches is required for the top guide to clear the top of the fuel channels. Without the repair, the licensee calculated that the maximum vertical separation would be 6.3 inches during a MSLB, assuming 360 degree through-wall weld failure of the H3 weld location (Reference 22). With the repair installed, the maximum vertical separation during a MSLB is limited to

0.056 inches at the H6 location, assuming 360 degree through-wall failure of any of the respective welds (Reference 5). This separation is limited by the tie-rods and should not impact the core spray system. ComEd analyzed the effect of 360 degree through-wall cracks in horizontal welds during a MSLB plus a DBE. The licensee stated that this combination event would result in a maximum momentary separation at one tie-rod stabilizer assembly location (i.e., tipping of the shroud) of 0.320 inches at the H6 weld (Reference 5). In addition, the largest vertical separation was calculated to be 0.452 inches at the H7 location during a DBE (Reference 5). The staff acknowledges that the ECCS performance and control rod insertion should not be impacted by any of the cases of momentary separation. Therefore, based on this assessment, the staff concluded that postulated separation during a MSLB, a MSLB plus DBE, or DBE plus normal pressure event would not preclude any of the systems from performing their safety functions.

2.4.6 Conclusion

The staff has evaluated the licensee's safety evaluation of the consequences of the proposed core shroud repair. The staff has found that the proposed repair should not impact the ability to insert control rods, the performance of the ECCS, particularly the core spray system, or the ability to reflood and cool the core. The staff concluded that the proposed repair does not pose adverse consequences to plant safety and, therefore, plant operation is acceptable with the proposed core shroud repair installed.

2.5 Materials, Fabrication and Inspection Considerations

2.5.1 Materials and Fabrication

ComEd stated (Reference 3) that Type 316 or 316L austenitic stainless steel, Type XM-19 stainless steel and nickel-based (Ni-Cr-Fe) alloy X-750 materials were selected for the fabrication of core shroud tie rod stabilizer components. These materials have been used for a number of other components in the BWR environment and have demonstrated good resistance to stress corrosion cracking by laboratory testing and long-term service experience. Welding is not used in the fabrication and the installation of the core shroud tie rod stabilizer, thereby, minimizing its susceptibility to IGSCC. The springs, supports and some connecting components were made from alloy X-750. The alloy X-750 material was selected for these components because of the requirements of higher material strength and lower coefficient of thermal expansion than that of the core shroud material (Type 304 stainless steel). The tie rods in the stabilizer assemblies were made of Type XM-19 stainless steel in a solution annealed condition with a carbon content less than 0.04 percent. The remaining connecting components in the tie rod stabilizer assemblies were made from either Type 316 or 316L austenitic stainless steel with a carbon content not more than 0.02 percent.

ComEd selected Type XM-19 instead of Type 304 or 316 stainless steel for the fabrication of tie rods in the stabilizer assemblies because Type XM-19 material has higher resistance to sensitization, higher allowable stress and a slightly lower coefficient of thermal expansion which would increase the thermal pre-load. ComEd stated that Type XM-19 was extensively tested in the mid-1970's, with the results published in Reference 27. The test results showed that Type XM-19 material has good resistance to sensitization and IGSCC. The solution annealed Type XM-19 material has been used in BWR environments with successful experience for over 20 years. The material was used for piston or index tubes in the control rod drive mechanisms and in a number of other applications.

Type 316 or 316L austenitic stainless steel and solution annealed alloy Type XM-19 are acceptable ASME Code Section III materials. The alloy X-750 was procured to American Society for Testing and Materials (ASTM) Standard B637, Grade UNS N07750 material (bars and forging) requirements. The heat treatment of alloy X-750 includes solution annealing at 1975 degrees Fahrenheit ±25 degrees Fahrenheit for 60 to 70 minutes, followed by forced air cooling, and age hardening at 1300 degrees Fahrenheit ± 15 degrees Fahrenheit for a minimum of 20 hours, followed by air cooling. The equalization heat treatment at 1500 degrees Fahrenheit to 1800 degrees Fahrenheit was prohibited because this heat treatment will produce a microstructure that would make the alloy X-750 material susceptible to IGSCC.

Type 316 or 316L austenitic stainless steel was procured to ASTM A-479, A-182 or A-240 with a maximum carbon content of 0.020 percent. The procured materials were water quenched from solution annealing at 2000 degrees Fahrenheit ±100 degrees Fahrenheit. ComEd stated that all Type 316 or 316L components were re-solution annealed and sensitization tested after final machining with the exception of electrolyzed (hard chrome plated) locking pins and the lower contact spacer.

The Type XM-19 stainless steel materials were procured to ASTM specification A182, A240, A412 or A479. The materials were solution annealed at 1950 degrees Fahrenheit to 2050 degrees Fahrenheit, followed by forced air cooling to a temperature below 500 degrees Fahrenheit in 20 minutes or less. The staff finds that the process of air-cooling from the solution annealing temperature is not consistent with the Boiling Water Reactor Vessel and Internals Project (BWRVIP) guidelines as provided in Reference 28, where water quenching from the solution annealing temperature is specified. ComEd stated that due to the straightness requirement in the fabrication of the tie rods, it is necessary to air cool the XM-19 materials from the solution annealing temperature, because water quenching will cause excessive distortion in the materials. To support the use of air cooled XM-19 material, ComEd submitted (Reference 5) a GE report of evaluating the stress corrosion cracking of XM-19 in the BWR environment. GE's evaluation report presented several sensitization and stress corrosion studies on XM-19 and several 300 series stainless steels with various carbon contents. The results of the studies had shown that, due to its sluggish kinetics of sensitization, XM-19 exhibited good resistance to sensitization and ranked very high in stress corrosion resistance among all the 300 series stainless steels tested. Based on the test data presented in Reference 5, the staff has determined that the air cooling rate specified in the fabrication of tie rods will not cause any sensitization in the XM-19 material. Therefore, the staff concludes that the subject air cooled XM-19 material is acceptable for use in the BWR environment.

All procured XM-19 and Type 316 or 316L stainless steel materials were tested for sensitization in accordance with ASTM Standard A262, Procedures A or E, to ensure the materials were not sensitized. These materials were also sensitization tested after high temperature annealing during fabrication. The maximum hardness of the procured materials and completed parts were specified in the GE Fabrication Specification (25A5690, Revision 2). The threaded areas of Type XM-19 tie rod stabilizer assembly components were re-solution annealed after final machining to remove the surface cold work effect. The cold work resulting from machining is known to promote IGSCC. ComEd stated that the re-solution annealing was carried out by induction heating at a frequency of approximately 8 khz, and that the induction heating process was qualified using heat treated 316L stainless steel threaded sections. GE has performed metallographic examination of the induction heated pieces. The result of the examination showed that a very thin machined skin layer on the threads was completely recrystallized and that a limited grain growth from an original grain size of 9 to 7.5 to 6 had occurred.

To preclude intergranular attack (IGA) as a result of high temperature annealing, ComEd required IGA testing per GE E50YP11 specification to be performed for each heat and heat treat lot of materials after annealing or pickling. In lieu of IGA testing, a minimum of 0.03 inches may be removed from all surfaces after the last exposure to high temperature annealing as a control of IGA.

ComEd indicated that tie rod stabilizer assembly components are generally rough machined to within 0.10 inch of final size and skim passes are used to achieve the final dimensions. Coolant and sharp tools were used in the machining. The final machined surface finish is generally specified to be 125 root mean square or better. ComEd also indicated that a Nickel-Graphite antiseize thread lubricant (D50YP5B) will be used in the installation of tie rod stabilizer assemblies. Controls of lubricant impurities were provided in the GE Specification (D50YP12), where impurities limits were specified for halogens, sulfur and nitrates. ComEd stated that machined components that were not solution annealed after machining, were metallographic and microhardness evaluated on test samples to verify that the surface condition after final machining has very shallow cold work depth. The acceptance criteria for machined surfaces were specified in GE's fabrication specification (25A5690, Revision 2).

The staff has reviewed ComEd's submittal regarding the proposed core shroud repair and concludes that the selected materials and fabrication methods for the tie rod stabilizer assemblies are acceptable.

2.5.2 Pre-Modification and Post-Modification Inspection

ComEd's pre-modification inspection plan (Reference 2) for Dresden, Unit 2, to support the repair installation consisted of inspection of vertical welds, ring segment welds, H-8 and H-9 welds and repair attachment locations, and was reviewed by the staff. The selection of the welds and the scope and limitation of the inspection are briefly summarized below. ComEd stated that the inspection plan for Dresden, Unit 3, will be submitted at a later date to support its fourteenth refueling outage, which is scheduled for the Fall of 1996.

(1) Ultrasonic examination (UT) was performed on seven (7) vertical welds (V14 through V19 and V28) of the core shroud, using the GE area scanner system. V14 through V19 welds are vertical welds between each pair of the horizontal welds of H3/H4 and H4/H5, and V28 weld is the vertical weld between horizontal welds H6/H7. The UT area scanner consisted of three transducers (45 degree shear, 60 degree RL and surface creeping wave). About 30 percent to 50 percent of each vertical weld (approximately 27 inches) was examined.

(2) Enhanced visual examination was performed on the remaining five (5) vertical welds (V5, V6, V7, V26 and V27) from the outside diameter (OD) surface as the inside diameter (ID) surface is not accessible. About 43 percent to 72 percent of each vertical weld (approximately 24 inches) was examined.

(3) Enhanced visual examination was performed on each segment weld of the shroud head flange ring (4 welds), top guide support ring (6 welds) and the core plate support ring (6 welds). Approximately twelve (12) inches of each segment weld was inspected.

(4) Enhanced visual examination was performed on the H-8 weld from the jet pump annulus region at the four repair assembly locations (20 degree, 110 degree, 200 degree and 290 degree Azimuth). The H-8 weld connects the core shroud support plate to the core shroud support ring. Approximately twelve (12) inches of H-8 weld at each repair location were inspected.

(5) Enhanced visual examination was performed on the H-9 weld from the jet pump annulus region at the four repair assembly locations (20 degree, 110 degree, 200 degree and 290 degree Azimuth). The H-9 weld connects the core shroud support plate to the reactor vessel. Approximately 12 inches of H-9 weld at each repair location were inspected.

(6) Enhanced visual examination was performed on all repair assembly attachment areas at four locations (20 degree, 110 degree, 200 degree and 290 degree Azimuth) before and after cutting or polishing operations. Each end of the four tie rod stabilizer assemblies was attached at the core shroud head flange and the core shroud support plate, respectively.

ComEd completed the above examinations on August 18, 1995, and reported the inspection results (Reference 29). ComEd stated that the ultrasonic examination and enhanced visual examination were performed in accordance with the BWRVIP guidelines provided in "Standards for Ultrasonic Examination of Core Shroud Welds" and "Standards for Visual Inspections of Core Shroud," respectively, and that no reportable indications were identified in area of interest. Because of the smooth machined surface condition, eddy current test was used in identifying the segment welds in the core shroud head flange ring, top guide support ring and core plate support ring.

ComEd reported (Reference 29) that the following circumferential cracking indications associated with the H3 and H5 welds were identified during the visual examination of the ring segment welds: (a) an indication approximately 2 inches long is located on the OD surface of the core plate support ring and is associated with the lower heat affected zone (HAZ) of H5 weld, (b) significant cracking approximately 60 inches in length is located on the ID surface of the top guide support ring and is predominantly associated with the upper HAZ of the H3 weld, and (c) some minor cracking (less than 12 inches) is located on the ID surface of the core shroud and is associated with the lower HAZ of the H3 weld. The reported circumferential cracking associated with horizontal welds H3 and H5 will not affect the structural integrity of the core shroud because welds H3 and H5 will be structurally replaced by the core shroud tie rod stabilizer assemblies.

ComEd has not yet finalized its reinspection plan for the core shroud and the tie rod stabilizer assembly components. The staff recommends that ComEd's reinspection plan should consider the following (1) the plant specific repair design requirements, (2) the extent and the results of the baseline inspection performed during pre-modification inspection, (3) the threaded areas and the locations of crevices and stress concentration in the tie rod stabilizer assemblies, and (4) BWRVIP reinspection guidelines when they are established. ComEd is requested to submit the Dresden, Unit 2, reinspection plan for the core shroud and repair assemblies within 6 months after restart of Dresden Unit 2. The NRC staff will review ComEd's reinspection plans when submitted. Since the core shroud and the tie rod stabilizer assemblies are generally classified as ASME Code Class B-N-2 components (core structural support), the reinspection plan will be required to be incorporated into the plant in-service inspection (ISI) program after NRC approval.

The staff has reviewed ComEd's pre-modification inspection plan and results. The staff concludes that the inspection performed by ComEd is acceptable to support the planned core shroud repair.

3.0 CONCLUSION

The proposed core shroud repair has been designed as an alternative to the requirements of the ASME Code, Section XI, pursuant to 10 CFR 50.55a(a)(3)(i). Based on a review of the core shroud modification hardware from structural, systems, materials, and fabrication considerations, as discussed above, the staff concludes that the proposed modifications of the Dresden, Units 2 and 3, core shrouds are acceptable and, subject to the submittal of the inservice inspection program, will not result in any increased risk to the public health and safety.

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Date: December 6, 1995

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2. Letter from J. L. Schrage, ComEd, to the USNRC Document Control Desk, "Submittal of Core Shroud Inspection Plan for Dresden Unit 2," March 30, 1995.

3. Letter from J. L. Schrage, ComEd, to the NRC Document Control Desk, with proprietary attachments, May 24, 1995.

4. Letter from J. Stang, NRC, to D. L. Farrar, ComEd, "Request for Additional Information - Core Shroud Repair" TAC Nos. M91301 and M91302, July 26, 1995.

5. Letter from Peter L. Piet, ComEd, to the NRC Document Control Desk, "Response to NRC Staff Request for Additional Information (RAI)," with proprietary attachments and enclosures, August 14, 1995.

6. Letter from J. L. Schrage, ComEd, to the NRC Document Control Desk, "Design Basis Discrepancy Related to Core Shroud Seismic Calculations," September 5, 1995.

7. Letter from Peter L. Piet, ComEd, to the NRC Document Control Desk, "Core Spray Flaw Evaluations," September 25, 1995.

8. Letter from Peter L. Piet, ComEd, to the NRC Document Control Desk, "Hardware Seismic Design with Improved Tie-Rod and Shroud Weld Crack Equivalent Rotational Stiffness for Dresden and Quad Cities," October 2, 1995.

9. Dresden 1 and 2, Updated Final Safety Analysis Report (UFSAR), Chapter 3 and Appendix C, Revision 1, June 1992.

10. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, 1989 Edition.

11. GENE 771-81-1194, Revision 2, "Commonwealth Edison Company Dresden Nuclear Power Plant Units 2 & 3, Shroud and Shroud Repair Hardware Analysis, Volume I, Shroud Repair Hardware."

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19. GE Stress Report, 25A5691, Revision 2, "Pressure Vessel - Dresden Units 2 & 3."

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21. Design Record File (DRF) for the Dresden Shroud Repair Program DRFB13-0749.

22. Letter from P. L. Piet, ComEd, to the NRC Document Control Desk, with attachments, December 14, 1994.

23. Letter from J. L. Schrage, ComEd, to the NRC Document Control Desk, with enclosure, July 10, 1995.

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25. Letter from P. L. Piet, ComEd, to the NRC Document Control Desk, with attachment, September 8, 1995.

26. GENE 771-44-0894, Revision 2, "Justification of Allowable Displacement of the Core Plate and Top Guide Core Shroud Repair," November 16, 1994 (Proprietary).

27. GE Document NEDE-21653-P, "XM-19 Materials Qualification Report," July 1977 (Proprietary).

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UNITED STATES
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WASHINGTON, D.C. 20555-0001

August 16, 1995

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SUBJECT: REQUEST FOR RELIEF RELATED TO THE THIRD TEN-YEAR INTERVAL ISI PROGRAM - DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3 (TAC NOS. M92421 AND M92422)

Dear Mr. Farrar:

By letter dated February 24, 1994, as supplemented by letter dated April 6, 1994, Commonwealth Edison Company (the licensee) submitted Revision 3 to the third ten-year interval inservice inspection (ISI) program plan. Revision 3 requested the staff's review and approval of Relief Request PR-14, Revision 1, PR-18 and CR-17.

Our evaluation of relief requests PR-14 and PR-18 was forwarded to you by letters dated May 25, 1994, and July 1, 1994, respectively.

Relief request CR-17 requested approval for the implementation of the alternative rules of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Section XI, Code Case N-524 dated August 9, 1993, entitled, "Alternative Examination Requirements for Longitudinal Welds in Class 1 and 2 Piping Section XI Division 1," pursuant to 10 CFR 50.55a(a)(3) to be applied to the ISI program for Dresden, Units 2 and 3.

The staff has reviewed relief request CR-17. The staff's evaluation and conclusions are contained in the enclosed Safety Evaluation (SE). The staff has concluded that the licensee's proposed alternative use of Code Case N-524 for Dresden, Units 2 and 3, is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) as compliance with the specified requirements of Section XI would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Use of Code Case N-524 is authorized until such time as this Code case is published in a future revision of Regulatory Guide 1.147. At that time if you intend to continue to implement this Code case, you are to do so by incorporating any limitations issued in Regulatory Guide 1.147.

Sincerely,

Robert A. Capra

Robert A. Capra, Director
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

AA3

Docket Nos. 50-237, 50-249

Enclosure: Safety Evaluation

cc w/encl: see next page

DFol 1/1

D. L. Farrar
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August 16, 1995

SUBJECT: REQUEST FOR RELIEF RELATED TO THE THIRD TEN-YEAR INTERVAL ISI PROGRAM - DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3 (TAC NOS. M92421 AND M92422)

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Sincerely,

Original signed by:

Robert A. Capra, Director
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-237, 50-249

Enclosure: Safety Evaluation

cc w/encl: see next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE INSERVICE INSPECTION PROGRAM REQUESTS FOR RELIEF FOR

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

DOCKET NOS. 50-237 AND 50-249

1.0 INTRODUCTION

By letter dated February 24, 1994, as supplemented April 6, 1994, Commonwealth Edison Company (ComEd, the licensee) requested approval for the implementation of the alternative rules of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Section XI, Code Case N-524 dated August 9, 1993, entitled, "Alternative Examination Requirements for Longitudinal Welds in Class 1 and 2 Piping Section XI, Division I," pursuant to 10 CFR 50.55a(a)(3) to be applied to the third ten year Inservice Inspection (ISI) program for Dresden, Units 2 and 3.

The Technical Specifications (TS) for Dresden Nuclear Power Station, Units 2 and 3, state that the inservice inspection and testing of the ASME Code Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Code and applicable Addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). In 10 CFR 50.55a(a)(3) it states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination on requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first ten-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) on the date twelve months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The applicable edition of Section XI of the ASME Code for the Dresden Nuclear Station, Units 2 and 3,

Enclosure

third 10-year ISI interval is the 1986 Edition. The components (including supports) may meet the requirements set forth in subsequent editions and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein and subject to Commission approval. However, the licensee has prepared the third ten-year interval inservice inspection program plan for Dresden Nuclear Power Station, Units 2 and 3, to meet the requirements of the 1989 Edition of the ASME Code. The third ten year interval inservice inspection program plan was approved by the NRC staff on May 19, 1994.

2.0 RELIEF REQUEST CR-17

2.1 Compound Identification

Longitudinal welds in Class 1 and 2 piping.

2.2 ASME Code, Section XI, Third Interval Requirements

Table IWC-1 Category C-F that require volumetric and/or a surface examination of longitudinal welds.

2.3 Licensee's Basis For Relief

The licensee states:

> Unlike circumferential welds, longitudinal welds are typically fabricated during original manufacturing under controlled shop conditions. In addition, the vast majority of longitudinal piping welds undergo solution heat treatment as part of the manufacturing process. Heat treatment enhances the material properties of the weld and reduces the residual stresses created by welding. Heat treatment of the piping and longitudinal weld also makes the material properties more uniform throughout the piping.
>
> The benefits of the enhanced material properties of shop fabricated longitudinal welds are demonstrated by the past 20 years of industry experience. In a survey conducted by the ASME Task Group on ISI Optimization it was found that the number of recordable indications discovered in longitudinal piping welds during 261 cumulative years of operation was very minimal. And more importantly, <u>none</u> of the recordable indications were found to be rejectable service induced flaws.
>
> On the basis of the above information, the additional costs and man-rem exposure associated with the incremental inspection of such welds, in association with circumferential butt weld inspections as currently required by Section XI, are not technically warranted. The ASME Code has recognized this fact and has recently published Code Case N-524 to allow alternate examination coverage of longitudinal piping welds.

Based on the above, Dresden Station requests relief from the current ASME Section XI requirements for examination coverage of longitudinal piping welds as specified in Tables IWB-2500-1 and IWC-2500-1.

2.4 Alternate Testing

The licensee proposes to apply Code Case N-524 as alternative rules for the examination of longitudinal welds in Class 1 and 2 piping.

2.5 Evaluation/Conclusions

The ASME Code, Section XI (1989 Edition), requires one pipe diameter in length, but no more than 12 inches, be examined for Class 1 longitudinal piping welds. Class 2 longitudinal piping welds are required to be examined for a length of 2.5t, where t is the thickness of the weld. These lengths of weld are measured from the intersection of the circumferential weld and longitudinal weld. The licensee's proposed alternative, Code Case N-524, limits the volumetric and surface examination requirements of the longitudinal weld to the volume or area contained within the examination requirements of the intersecting circumferential weld.

Longitudinal welds are produced during the manufacturing process of the piping, not in the field - as is the case for circumferential welds. The ASME Code contains requirements for characteristics and performance of materials and products, and specifies examination requirements for the manufacturing of the subject longitudinal piping welds.

In addition, there are material, chemical, and tensile strength requirements in the Code. The manufacturing process that is specified by the Code provides assurance of the structural integrity of the longitudinal welds at the time the piping is manufactured.

The preservice examination and subsequent inservice examinations have provided assurance of the structural integrity of the longitudinal welds during the service life of the plant to date. The experience in the United States has been that ASME Code longitudinal welds have not experienced degradation that would warrant continued examination beyond the boundaries required to meet the circumferential weld examination requirements. No significant loading conditions or known material degradation mechanisms, which specifically relate to longitudinal seam welds in nuclear plant piping, have become evident to date. If any degradation associated with a longitudinal weld were to occur, it is expected that it would be located at the intersection with a circumferential weld. This intersection is inspected in accordance with the provisions of Code Case N-524. In addition, there is a significant accumulation of man-rem associated with the examination of longitudinal welds, especially in Class 1 piping. The staff concludes that continued imposition of the Code examination requirements for longitudinal welds constitutes a hardship without a compensating increase in the level of quality and safety.

3.0 CONCLUSION

Accordingly, the licensee's proposed alternative to use Code Case N-524 is authorized for Dresden Nuclear Station, Units 2 and 3, pursuant to 10 CFR 50.55a(a)(3)(ii) until such time as the Code Case is published in a future revision of Regulatory Guide 1.147. At that time, the licensee is to follow all provisions in Code Case N-524, with limitations issued in Regulatory Guide 1.147, if any, if the licensee continues to implement this relief request.

Principal Contributor: John Stang

Date: August 16, 1995

[redacted]

50-237/249

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 2, 1995

Mr. D. L. Farrar
Manager, Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: REQUEST FOR WITHHOLDING INFORMATION FROM PUBLIC DISCLOSURE

Dear Mr. Farrar:

By letter from Commonwealth Edison Company (ComEd) dated June 7, 1995, and General Electric Company's (GE) affidavit executed by George B. Stramback dated June 1, 1995, you submitted a proprietary document entitled, "Transmittal of Computer Runs for Shroud Repair Seismic Analysis," dated June 1, 1995, with attached computer runs 2788T, 2794T, 2790T and 2466T, dated April 1995, and requested that it be withheld from public disclosure pursuant to 10 CFR 2.790.

GE stated that the information should be considered exempt from mandatory public disclosure for the following reasons:

"(4)a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

(4)b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

(8) The information identified ... above, is classified as proprietary because it contains detailed results of analytical models, methods and processes, including computer codes, and it contains the supporting Design Record File (DRF) detailed calculations, results and bases for conclusions. These reports are part of the DRF supporting information to evaluate a hardware design modification (stabilizer for the shroud horizontal welds) intended to be installed in a reactor to resolve the reactor pressure vessel core shroud weld cracking concern. This detailed level of information

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usually resides in GENE files, only for audit by customers and the NRC. This information shows in specific detail the processes, codes and methods employed to perform the evaluations summarized in the above identified document. * * *"

We have reviewed your submittal and the material in accordance with the requirements of 10 CFR 2.790 and, on the basis of GE's statements, have determined that the submitted information sought to be withheld contains trade secrets or proprietary commercial information.

Therefore, we have determined that the document entitled "Transmittal of Computer Runs for Shroud Repair Seismic Analysis," dated June 1, 1995, with attached computer runs 2788T, 2794T, 2790T and 2466T, dated April 1995, marked as proprietary will be withheld from public disclosure pursuant to 10 CFR 2.790(b)(5) and Section 103(b) of the Atomic Energy Act of 1954, as amended.

Withholding from public inspection shall not affect the right, if any, of persons properly and directly concerned to inspect the document. If the need arises, we may send copies of this information to our consultants working in this area. We will, of course, insure that the consultants have signed the appropriate agreements for handling proprietary information.

If the basis for withholding this information from public inspection should change in the future such that the information could then be made available for public inspection, you should promptly notify the NRC. You should also understand that the NRC may have cause to review this determination in the future, for example, if the scope of a Freedom of Information Act request includes your information. In all review situations, if the NRC needs additional information from you or makes a determination adverse to the above, you will be notified in advance of any public disclosure.

Sincerely,

John F. Stang

John F. Stang, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-237, 50-249

cc: see next page

D. L. Farrar
Commonwealth Edison Company

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Sidley and Austin
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Site Vice President
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Mr. J. Heffley
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Chairman
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1320 Union Street
Morris, Illinois 60450

David J. Robare
General Electric Company
175 Curtner Avenue
San Jose, California 95125

usually resides in GENE files, only for audit by customers and the NRC. This information shows in specific detail the processes, codes and methods employed to perform the evaluations summarized in the above identified document. * * *"

We have reviewed your submittal and the material in accordance with the requirements of 10 CFR 2.790 and, on the basis of GE's statements, have determined that the submitted information sought to be withheld contains trade secrets or proprietary commercial information.

Therefore, we have determined that the document entitled "Transmittal of Computer Runs for Shroud Repair Seismic Analysis," dated June 1, 1995, with attached computer runs 2788T, 2794T, 2790T and 2466T, dated April 1995, marked as proprietary will be withheld from public disclosure pursuant to 10 CFR 2.790(b)(5) and Section 103(b) of the Atomic Energy Act of 1954, as amended.

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Sincerely,

Original signed by:

John F. Stang, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-237, 50-249

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50-237/249

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 28, 1995

Mr. D. L. Farrar
Manager, Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: REQUEST FOR WITHHOLDING INFORMATION FROM PUBLIC DISCLOSURE

Dear Mr. Farrar:

By letter from Commonwealth Edison Company (ComEd) dated May 24, 1995, and General Electric Company's (GE) affidavit executed by David Robare, dated May 19, 1995, you submitted proprietary documents entitled, "Back-Up Calculation for RPV Stress Report No. 25A5691, Dresden Units 2 & 3," GENE-771-77-1194, Revision 2; "Backup Calculations for Dresden Shroud Repair, Shroud Stress Report, Volume II, Dresden Units 2 & 3," GENE-771-82-1194, Revision 1; "Shroud and Shroud Repair Hardware Analysis, Shroud Repair Hardware Backup Calculations, Dresden Units 2 & 3," GENE-771-83-1194, Revision 1; "Shroud Repair Seismic Analysis, Dresden Units 2 & 3," GENE-771-84-1194, Revision 2; "Shroud Repair Seismic Analysis Backup Calculations, Dresden Units 2 & 3," GENE-771-85-1194, Revision 2; "Top Ring Plate and Star Truss Stress Analysis Backup Calculations, Dresden Units 2 & 3," GENE-771-96-0195, Revision 1; "Dresden Units 2 & 3, Primary Structure Seismic Models," GENE-523-A181-1294, Revision 1, December 1994, and requested that they be withheld from public disclosure pursuant to 10 CFR 2.790.

GE stated that the information should be considered exempt from mandatory public disclosure for the following reasons:

"(4)a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

(4)b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

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(8) ...it contains detailed results of analytical models, methods and processes, including computer codes, and it contains the supporting Design Record File (DRF) detailed calculations, results and bases for conclusions. These reports are part of the DRF supporting information to evaluate a hardware design modification (stabilizer for the shroud horizontal welds) intended to be installed in a reactor to resolve the reactor pressure vessel core shroud weld cracking concern. This detailed level of information usually resides in GENE files, only for audit by customers and the NRC. This information shows in specific detail the processes, codes and methods employed to perform the evaluations summarized in the above identified document...."

We have reviewed your submittal and the material in accordance with the requirements of 10 CFR 2.790 and, on the basis of GE's statements, have determined that the submitted information sought to be withheld contains trade secrets or proprietary commercial information.

Therefore, we have determined that the documents entitled "Back-Up Calculation for RPV Stress Report No. 25A5691, Dresden Units 2 & 3," GENE-771-77-1194, Revision 2; "Backup Calculations for Dresden Shroud Repair, Shroud Stress Report, Volume II, Dresden Units 2 & 3," GENE-771-82-1194, Revision 1; "Shroud and Shroud Repair Hardware Analysis, Shroud Repair Hardware Backup Calculations, Dresden Units 2 & 3," GENE-771-83-1194, Revision 1; "Shroud Repair Seismic Analysis, Dresden Units 2 & 3," GENE-771-84-1194, Revision 2; "Shroud Repair Seismic Analysis Backup Calculations, Dresden Units 2 & 3," GENE-771-85-1194, Revision 2; "Top Ring Plate and Star Truss Stress Analysis Backup Calculations, Dresden Units 2 & 3," GENE-771-96-0195, Revision 1; "Dresden Units 2 & 3, Primary Structure Seismic Models," GENE-523-A181-1294, Revision 1, December 1994, marked as proprietary will be withheld from public disclosure pursuant to 10 CFR 2.790(b)(5) and Section 103(b) of the Atomic Energy Act of 1954, as amended.

Withholding from public inspection shall not affect the right, if any, of persons properly and directly concerned to inspect the document. If the need arises, we may send copies of this information to our consultants working in this area. We will, of course, insure that the consultants have signed the appropriate agreements for handling proprietary information.

If the basis for withholding this information from public inspection should change in the future such that the information could then be made available for public inspection, you should promptly notify the NRC. You should also understand that the NRC may have cause to review this determination in the future, for example, if the scope of a Freedom of Information Act request includes your information. In all review situations, if the NRC needs additional information from you or makes a determination adverse to the above, you will be notified in advance of any public disclosure.

Sincerely,

John F. Stang

John F. Stang, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-237, 50-249

cc: see next page

D. L. Farrar
Commonwealth Edison Company

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David J. Robare
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San Jose, California 95125

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Sincerely,

Original signed by:

John F. Stang, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

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July 26, 1995

Mr. D. L. Farrar, Manager
Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 OPUS Place, Suite 500
Downers Grove, IL 60515

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - CORE SHROUD REPAIR (TAC NOS. M91301 AND M91302)

Dear Mr. Farrar:

By letters dated May 24, June 6 and July 10, 1995, Commonwealth Edison Company (ComEd) submitted information to the NRC concerning the core shroud repair for the Dresden Nuclear Power Station, Units 2 and 3. On July 18, 1995, the staff held a conference call with ComEd and their consultants to discuss this repair. During the call, a list of preliminary questions were raised by the staff. Enclosed please find the Request for Additional Information (RAI) developed from this call. This information is required for the staff to complete the review of the Dresden core shroud repair.

Please provide this information as soon as possible to allow the staff to complete its review in a timely manner.

This requirement affects nine or fewer respondents and, therefore, is not subject to Office of Management and Budget review under P.L. 96-511.

If you have any further questions, please contact me at (301) 415-1345.

Sincerely,

Original signed by

John Stang, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-237, 50-249

Enclosure: Request for Additional Information

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D. L. Farrar
Commonwealth Edison Company

Dresden Nuclear Power Station
Unit Nos. 2 and 3

cc:

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Springfield, Illinois 62704

Chairman
Grundy County Board
Administration Building
1320 Union Street
Morris, Illinois 60450

REQUEST FOR ADDITIONAL INFORMATION
CORE SHROUD REPAIR
DRESDEN, UNITS 2 AND 3

(1) In your design specification (25A5688, Revision 2), Sections 4.4.3 and 4.7, welding is identified as a repair contingency for austenitic 300 series stainless steel and in Section 4.4.3, assembly welds were mentioned. Please identify under what conditions repair welding and assembly welds will be applied during the fabrication and installation of the core shroud repair components. What are the controls or mitigation methods that will be implemented to minimize the magnitude of the residual stresses and material sensitization when applying welding?

(2) BWRVIP has issued the following documents to provide guidelines for visual examination (VT) and ultrasonic examination (UT) of core shrouds: (a) Standards for Visual Inspection of Core Shrouds, and (b) Core Shroud NDE Uncertainty & Procedure Standard. The guidelines in these documents should be followed in the examination of the core shroud and repair assemblies. If you do not intend to reference the subject BWRVIP documents in your examination specifications or procedures, please identify all the exceptions you are going to take against the referenced BWRVIP guidelines.

(3) When detailed heat treatment records (time, temperature and cooling rate) are not available, what kind of testing do you perform to ensure that the fabricated alloy X-750 components are properly heat treated?

(4) General Electric stated in their fabrication specification, 25A5690, Revision 2, Section 3.2, that critical, highly stressed, machined areas such as the tie rod threads (XM-19) will be resolution annealed after machining to remove a possible cold worked layer.

(a) Please describe the resolution annealing process and provide details regarding how this process was qualified and the results of your metallurgical evaluation of the tie rod threads after resolution annealing such as its effect on the material hardness, grain sizes, surface oxidation and the state of sensitization. If the qualification was not performed on XM-19 materials, please justify why a similar qualification process need not be applied to XM-19 materials.

(b) General Electric stated that a minimum of 0.030 inches of austenitic 300 series and XM-19 stainless steel and alloy X-750 materials may be removed after high temperature annealing as a control of intergranular attack (IGA). Please provide the test data to support that the removal of 0.030 inches of surface material would effectively eliminate the IGA effect resulting from all high temperature annealing.

(c) In Section 3.2.2.1 it was stated that the electrolyzing process (hard chrome plating) will be applied to the locking pins after centerless grind to size. Please describe how this process was qualified and its controlling parameters established. What is the required quality control testing to ensure the plating has correct thickness and acceptable surface condition (no surface defect in the plating or pitting in the base metal)?

(5) Please identify all the threaded areas and locations of crevices and stress concentration in each component of the core shroud repair assemblies. In the planning of in-service inspection those areas should be emphasized for inspection because these areas are most susceptible to stress corrosion cracking. Please provide these information in tables and supplement it with sketches.

(6) Please provide details of your controls in the practices of machining, grinding and threading to minimize the effect of cold work, such as amount of materials to be removed in each pass, application of coolant and sharpness of the tool.

(7) The staff realizes that the repair assemblies may be inspected by a combination of visual and ultrasonic examinations. However, the staff has some concerns regarding the reliability of such inspection to identify the potential degradation in the threaded joints and areas of crevices and stress concentration, which have limited access for inspection. Please provide a discussion and/or propose an alternative inspection such as disassembling the threaded joints for inspection to ensure that the areas mentioned above in the repair assemblies will be adequately inspected for early detection of potential degradation.

(8) Please provide details of your planned baseline in-service inspection (location, extent, frequency, methodology and justification) of the core shroud to support the core shroud repair.

(9) Please provide details of your planned in-service inspection (location, extent, frequency, methodology and justification) of the installed core shroud repair components. Your planned inspection should consider the staff recommendation in Item 7.

If complete information for Items 5 and 9 can not be provided at this time, identify the date when such information will be provided.

(10) Please identify the lubricants that would be used on the machined threads during installation. What are the controls of the content of chlorides, sulfides, halogens and other elements that are known to promote stress corrosion cracking in stainless steel and high nickel alloy?

(11) Please discuss how are you going to monitor the magnitude of the spring preload to ensure there is no substantial relaxation of the preload. Please also discuss the safety consequences if the spring preload is completely relaxed and the feasibility of measuring the overall preload during plant operation.

(12) In your shroud and shroud repair hardware stress analysis (GENE-771-81-1194, Revision 2), Section 3.2, tie rods are specified to be made of XM-19 material.

(a) Please discuss the reasons for selecting XM-19 material instead of austenitic 304 or 316 stainless steel (low carbon content), and provide the relevant service experience and laboratory testing data to support its application in the BWR environment.

(b) It should be noted that the acceptable yield strength of XM-19 material is limited to 90 ksi. Is this upper limit of the yield strength for XM-19 identified in your procurement specification?

(c) The staff finds that your specified heat treatment of air-cooling from the solution annealing temperature for XM-19 materials is not consistent with the BWRVIP guidelines provided in the document (BWROG-VIP-9410) of "BWR Core Shroud Repair Design Criteria," where water quenching from the solution annealing temperature is recommended. Since there is very limited service experience of XM-19 material in the BWR environment, the staff recommends that an accelerated stress corrosion testing of a mock-up simulating the XM-19 tie rod thread joint in a BWR environment should be performed to ensure there is no development of unexpected degradation.

(13) If the credit for the fillet or any circumferential welds in the core shroud is taken in the design of the proposed repair to maintain the required preload, please discuss in detail and provide the justification regarding the measures you plan to take, such as inspection, to ensure the welds are, and remain, in the condition assumed in the analyses.

(15) In GENE 771-81-1194, Revision 1, Volume 1, "Shroud Repair Hardware," Figure 6.3.2, page 37 shows the deformed configuration of long upper supports. Clarify the boundary conditions applied to the finite element model at the interface between the long upper support, the shroud flange, and the shroud head flange.

(16) Provide the preload and gap calculations, similar to those provided for Quad Cities 1 and 2, in GENE-771-68-1094, Supplement A to Revision 4, April 1995.

(17) In GENE 771-84-1194, "Shroud Repair Seismic Analysis," (Enclosure 9) and GENE-523-A181-1294, "Primary Structure Seismic Models" (Enclosure 15), show the weights which form the basis for the masses in the model comprising the shroud.

(18) Provide an evaluation of the core spray piping for emergency and faulted loading combinations which include MSLB and RLB loads.

July 20, 1995

Mr. D. L. Farrar
Manager, Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 OPUS Place, Suite 500
Downers Grove, IL 60515

SUBJECT: PARTIAL IMPLEMENTATION OF AMENDMENT - DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3 (TAC NOS. M92043 AND M92044)

Dear Mr. Farrar:

The Commission issued Amendment Nos. 134 and 128 to Appendix A of Facility Operating Licenses DPR-19 and DPR-25 on June 13, 1995. By letter dated June 29, 1995, Commonwealth Edison Company informed the Commission that it would immediately implement a portion of the amendment relating to the new setpoint for the Condenser Low Vacuum SCRAM for Dresden Station, Units 2 and 3. The staff has reviewed the Technical Specification (TS) pages that include the revised setpoint. These TS pages should adequately control the partial implementation. It is the staff's understanding that no other portions of Amendment Nos. 134 and 128 will be implemented until full implementation of the Technical Specification Upgrade Program.

Sincerely,

Original signed by:

John F. Stang, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-237, 50-249

cc: see next page

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NAME	CMoore		JStang		RCapra Rc		
DATE	07/20/95		07/20/95		07/20/95		

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D. L. Farrar
Commonwealth Edison Company

Dresden Nuclear Power Station
Unit Nos. 2 and 3

cc:

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July 18, 1995

Mr. John F. Opeka
Executive Vice President, Nuclear
Connecticut Yankee Atomic Power Company
Northeast Nuclear Energy Company
P.O. Box 270
Hartford, CT 06141-0270

SUBJECT: MILLSTONE NUCLEAR POWER STATION UNIT NO. 3 - SAFETY EVALUATION FOR TOPICAL REPORT, NUSCO-152, ADDENDUM 4, "PHYSICS METHODOLOGY FOR PWR RELOAD DESIGN," (TAC NO. M91815)

Dear Mr. Opeka:

By letter dated March 28, 1995, the Northeast Nuclear Energy Company (NNECO) submitted for staff review Topical Report NUSCO-152, ADDENDUM 4 "PHYSICS METHODOLOGY FOR PWR RELOAD DESIGN," regarding the use of the approved Westinghouse computer code package for Cycle 7 application.

The NRC staff reviewed the topical report and finds the use of the enhanced computer codes acceptable. A copy of the Safety Evaluation (SE) is enclosed. With the issuance of this SE, the staff considers TAC No. M91815 complete. If you have any questions or comments regarding the SE, please call me at (301)415-3045.

Sincerely,

Original signed by:

Vernon L. Rooney, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-213

Enclosure: Safety Evaluation

cc w/encl: See next page

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July 18, 1995

Mr. John F. Opeka
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SUBJECT: MILLSTONE NUCLEAR POWER STATION UNIT NO. 3 - SAFETY EVALUATION FOR TOPICAL REPORT, NUSCO-152, ADDENDUM 4, "PHYSICS METHODOLOGY FOR PWR RELOAD DESIGN," (TAC NO. M91815)

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Sincerely,

Original signed by:

Vernon L. Rooney, Senior Project Manager
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Docket No. 50-213

Enclosure: Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 18, 1995

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SUBJECT: MILLSTONE NUCLEAR POWER STATION UNIT NO. 3 - SAFETY EVALUATION FOR TOPICAL REPORT, NUSCO-152, ADDENDUM 4, "PHYSICS METHODOLOGY FOR PWR RELOAD DESIGN," (TAC NO. M91815)

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Sincerely,

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Docket No. 50-213

Enclosure: Safety Evaluation

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO TOPICAL REPORT NUSCO-152, ADDENDUM 4

"PHYSICS METHODOLOGY FOR PWR RELOAD DESIGN"

NORTHEAST UTILITIES SERVICE COMPANY

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

DOCKET NO. 50-423

1.0 INTRODUCTION

In a submittal of March 8, 1995 (Ref. 1), the Northeast Utilities Service Company (NUSCO) requested review and approval of the topical report NUSCO-152, "Physics Methodology for PWR Reload Design, Addendum 4, January 3, 1995" (Ref. 2). The report described the use of an approved Westinghouse (W) methodology and computer code package for Millstone Unit 3, beginning with the Cycle 7 reload design. This report documents the capability of NUSCO to perform in-house core reload nuclear design analyses for Millstone Unit 3 using standard W methodologies previously approved by the NRC.

NUSCO intends to use the currently approved W methodology and computer programs for Pressurized Water Reactor (PWR) reload applications, including steady-state reload physics design, calculations for startup predictions, generation of physics and kinetics input for transient and safety analyses and for the plant reactivity computer.

2.0 SUMMARY OF THE TOPICAL REPORT

This addendum to the topical report describes the enhanced W computer programs and physics models used by NUSCO to analyze reload cores and compares the model predicted results with measurements obtained from benchmarking data covering Millstone Unit 3 operating Cycles 3, 4, and 5. The Millstone Unit 3 analyses were performed over a range of conditions from hot zero power (HZP) to hot full power (HFP) operation. The agreement between the measured and calculated values presented in the topical report is used to validate the application of the computer programs for analysis of Millstone Unit 3.

NUSCO intends to use these methods for steady-state PWR core physics reload design applications, including fuel assembly and loading pattern analysis, startup predictions, and safety analysis inputs.

Enclosure

2.1 Overview

Section 1 of the topical report provides introductory background information and an overview of the objectives and scope of the report.

2.2 Physics Codes

Section 2 of the topical report provides a description of each of the individual computer codes. The major W codes used by NUSCO are PHOENIX-P (Ref. 3), ANC (Ref. 4), FIGHT-H (Ref. 5, 6), and APOLLO (Ref. 7).

2.3 Physics Methodology

Section 3 of the topical report describes the approved W PWR methodology used by NUSCO, and outlines the procedures used for the model applications.

2.4 Physics Model Applications

Section 4 of the topical report describes the application of the previously specified Westinghouse physics methodology in four major areas:

- core power distributions at steady-state conditions,
- axial power distribution control limits,
- core reactivity parameters, and
- core physics parameters for transient analysis input.

2.5 Physics Model Verification

Section 5 of the topical report describes three operating cycles of Millstone Unit 3 which provided measured plant data from a range of plant startup and normal operation conditions. Millstone Unit 3 is a four-loop W PWR plant with a 17x17 fuel rod array, 193 fuel assembly core, generating 3411 megawatts-thermal (Mwt) at rated power, which began commercial operation in 1986. There are 61 full-length rod cluster control assemblies (RCCAs). The in-core flux instrumentation consists of moveable fission chambers which can be inserted into multiple core locations. The neutron flux detector signals are processed off-line with the W INCORE program (Ref. 8) to infer the 3D measured power distribution in the core.

The topical report compares the calculated PWR physics parameters with measured or inferred plant data. The measured data cover the range from zero power startup testing to normal full power operations. Three operating cycles were included.

The key PWR physics parameters for which comparisons of predicted to measured or inferred plant data were performed to provide verification of NUSCO's ability to apply the W methodology to plant-specific reload designs are listed. The parameters measured during zero physics tests are:

- critical boron concentration,
- isothermal temperature coefficient, and
- control rod worth.

For each of the parameters compared, the observed differences were compared to a set of startup test review criteria which represent the maximum expected deviation between prediction and measurement (Ref. 9).

The parameters measured or inferred during at power operation include:

- boron letdown curves,
- power peaking factors, F_Q and $F_{\Delta H}$,
- radial power distributions,
- axial power distributions, and
- axial offset.

3.0 TECHNICAL EVALUATION

3.1 Background

NUSCO has been a technology licensee of W since 1985, through which the relevant physics design methodology and associated computer programs have been obtained, beginning in 1986. The licensee states that all methods employed and described in this topical report (including model development, computer programs, measured data processing, etc.) are standard W methods and reflect current practices. NUSCO has used the W methodology to model operating Cycles 3 through 5, and has performed detailed comparisons of the results to measured operating data. An evaluation of these comparisons is presented below for the key PWR physics parameters to be generated by the licensee.

3.2 Critical Boron Concentrations

Critical boron concentrations (CBC) were measured at HZP conditions with all rods out (ARO) and with banks D, C, B and A fully inserted. The ANC 3D model predictions of CBC were compared to zero-power startup test measurements as well as W ANC predictions. All differences between calculated and measured boron ppm data are within the physics test review and acceptance criterion of ± 50 ppm. The results from the HZP comparisons qualify the model for predicting the CBC and core reactivity for beginning-of-cycle (BOC), xenon-free conditions.

3.3 Isothermal Temperature Coefficient

The isothermal temperature coefficient (ITC) is defined as the change in reactivity due to an incremental change in the core average moderator and fuel temperature. Measured ITCs were compared for both rodded and unrodded conditions to NUSCO and W ANC model predictions. All differences between

NUSCO ANC predictions and measured data are within the physics test acceptance criterion of ± 2 pcm/°F from the three cycles of operation. Note that 1 pcm is equivalent to $1x10^{-5}$ percent delta-K/K.

3.4 Control Rod Worths

Control rod worth is the reactivity difference (pcm) between different control rod configurations. The worth of the control rod banks A, B, C, and D was measured by boron dilution, using step-wise bank insertion and summing the differential worths obtained from the reactivity computer. The 3D ANC model was used for the prediction of the individual control rod bank worths and was compared with the BOC zero-power startup measurements for three operating cycles. All differences between NUSCO ANC predictions and measured bank worths are within the test review criteria of ±15% or 100 pcm, whichever is greater.

3.5 Radial Power Distributions

The measured radial power distributions are inferred by the INCORE procedure, after the flux map measurements are performed using the moveable incore neutron flux detector system. The predicted power distributions from the 3D ANC calculations are compared to measured values at several burnup intervals. The predictions show good agreement with the average difference between measured and predicted assembly powers less than 1.67% with a standard deviation less than 1.25%.

3.6 Axial Power Distributions and Axial Offset

A total of 12 axial power distribution measurements from the above flux maps over the three cycles of operation were plotted with the 3D ANC model predicted values at similar depletion points. The measured axial offset (AO), defined as the percent difference between the relative power in the top half of the core and that in the bottom half of the core, is also inferred by INCORE and is compared with the predicted values from ANC at 25 flux map statepoints. In general, the overall agreement between measured and predicted values of axial power distribution and axial offset are good. A larger than expected disagreement was observed during the latter part of Cycle 4 and has been attributed to plate out of soluble boron in certain areas of the core. Since the W predicted axial power shapes are essentially identical to the NUSCO predicted values, this tends to confirm that the Cycle 4 disagreements are due to this unusual physical phenomenon and its effect on the measurements.

3.7 Power Peaking Factors

Measured values of the primary power peaking factors, the heat flux hot channel factor (F_Q) and the nuclear enthalpy rise hot channel factor ($F_{\Delta H}$), were inferred using the W INCORE program. The predicted power peaking factors were obtained from the 3D ANC model depletion results at the closest burnup intervals. For F_Q, the largest absolute difference between the measured and predicted values for 25 measured statepoints over the three cycles was 7.1% and occurred in Cycle 4 due to the axial anomaly mentioned previously. For Cycles 3 and 5, the agreement was much better, the largest difference being 4.8%. For $F_{\Delta H}$, the largest absolute difference was 2.8%.

3.8 Boron Rundown Curves

Critical boron concentrations from measured HFP, equilibrium xenon and samarium conditions were compared to both W and NUSCO 3D ANC model predicted boron rundown curves for three operating cycles. NUSCO and W predictions are generally identical and the measurements from three operating cycles, taken at the time of INCORE power distribution measurements, show good agreement with predicted values.

4.0 SUMMARY AND CONCLUSIONS

The licensee has performed substantial benchmarking using currently accepted W reload design methodologies. This effort consisted of detailed comparisons of the calculated physics parameters with the measurements obtained from operating Millstone Unit 3 as well as with W predictions. In general, the NUSCO ANC predictions agreed well with measurements. All startup test predictions fell within the required review and acceptance criteria. In addition, comparisons between power operation measurements and NUSCO ANC predictions for boron rundown, peaking factors, and power distributions show good agreement. This effort demonstrated the capability of NUSCO to use the W computer program package for application to Millstone Unit 3 using the W Relaxed Axial Offset Control (RAOC) power distribution control limit calculational procedure (Ref. 10).

Based on the analyses and results presented in the topical report, the staff concludes that the W methodology, as validated by NUSCO, can be applied to steady-state PWR reactor physics calculations for the Millstone Unit 3 reload design applications discussed in the above technical evaluation. The accuracy of this methodology has been demonstrated to be sufficient for use in design applications, including PWR reload physics analysis, generation of transient analysis inputs, startup predictions and plant reactivity computer inputs.

As in similar approvals, application of the approved package is to be limited to the fuel configuration and core design parameters verified in the topical report. Changes in the fuel vendor or introduction of significantly different fuel designs may require further validation by the licensee.

5.0 REFERENCES

1. Letter from J. F. Opeka (NUSCO) to Document Control Desk (USNRC), "Millstone Nuclear Power Station, Unit No. 3, Physics Methodology for PWR Reload Design," dated March 8, 1995.

2. NUSCO-152, Addendum 4, "Physics Methodology for PWR Reload Design," Northeast Utilities Service Company, January 3, 1995.

3. WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," November 1987.

4. WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Program," December 1985.

5. WCAP-6073, "LASER - A Depletion Program for Lattice Calculations Based on MUFT and THERMOS," April 1966.

6. WCAP-2048, "The Doppler Effect for a Non-Uniform Temperature Distribution in Reactor Fuel Elements," July 1962.

7. WCAP-13524-P, "APOLLO - A One Dimensional Neutron Diffusion Theory Program," October 1992.

8. WCAP-8498, "INCORE Power Distribution Determination in Westinghouse Pressurized Water Reactors," July 1975.

9. ANSI/ANS-19.6.1 (1985) "Reload Startup Physics Tests for Pressurized Water Reactors".

10. WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control/FQ Surveillance Technical Specifications," June 1983.

Principal Contributor: L. Kopp

Date: July 18, 1995

Docket File

50-237

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 2, 1995

Mr. D. L. Farrar, Manager
Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

Dear Mr. Farrar:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RELATED TO THE PROPOSED USE OF THE CORPORATE EOF AS AN INTERIM EOF (TAC NOS. M84864, M84865, M84866, M84867, M84868, M84869, M84870, M84871, M84872, M84873, M84874, AND M84875)

On May 11, 1995, a conference telephone call was held between the NRC staff and members of the Commonwealth Edison Company (ComEd) staff during which we discussed the status of ComEd's proposal to staff and activate, within an hour following an emergency, an interim emergency operations facility (EOF) at the corporate offices until the near-site EOF is staffed and activated. As part of the discussions, the NRC staff informed ComEd of its remaining open issues and requested additional information. This letter formally requests ComEd to provide the requested additional information.

In NRC Inspection Report 50-237/92022, dated August 20, 1992, the NRC staff identified a concern that the ComEd emergency plan does not provide for timely augmentation of its emergency response organization to relieve the control room and technical support center of off-site emergency response functions. In response to this concern, ComEd proposed (in its September 1992 emergency plan revision) to activate the corporate EOF, within an hour following an emergency, as an interim EOF until the near-site EOF could be staffed. In the period since the submittal, NRC has held numerous meetings and conference calls with ComEd personnel. The NRC staff has observed the implementation of the Corporate EOF as an interim EOF during a number of emergency exercises and drills and has repeatedly identified concerns with ComEd's implementation of its proposed plan.

Earlier concerns focused on ComEd's inability to effectively transfer command and control between emergency response facilities and to adequately perform the functions of an EOF with the limited staff assigned to the Corporate EOF. As stated in the May 11, 1995, conference call, the staff observed the activation, staffing, and operation of the interim EOF during a Braidwood and a Zion exercise in April of this year. The staff's concerns regarding staffing and operation were resolved based upon their observations. The remaining concern is ComEd's inability to staff the Corporate EOF within one hour following an emergency declaration at one of its nuclear plants.

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Although there has been considerable emphasis on ComEd's capability to activate the Corporate EOF within about 1 hour, please note that even though ComEd has proposed to eliminate the need for prompt staffing of the near-site EOFs, the NRC has not approved this proposed change. A review of recent augmentation drills indicates that it has taken as long as 3½ hours to staff one of the near-site EOFs. Consequently, the need for timely staffing of the near-site EOFs remains. Until the staff approves the requested change to the interim EOF, it is expected that ComEd will use its best efforts to activate the near-site EOFs within the one hour goal.

Although we believe that a number of the concerns identified earlier have been resolved, progress toward resolution of the time for staffing, which was an initial staff concern in 1992, has been very slow. With this now as the remaining item, we request a response within 30 days of receipt of this letter so that we may complete our evaluation and promptly make our recommendation to the Commission. A summary of our outstanding concerns and request for additional information is enclosed. If there are questions regarding this request, please contact Mr. George Dick at (301) 415-3019.

This requirement affects nine or fewer respondents and, therefore, is not subject to Office of Management and Budget review under P.L. 96-511.

Sincerely,

Original signed by:

George F. Dick, Jr., Project Manager
Project Directorate III-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-456, STN 50-457, STN 50-454, STN 50-455, 50-237, 50-249, 50-373, 50-374, 50-254, 50-265, 50-295, 50-304

Enclosure: Request for Additional Information

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Byron, Illinois 61010

REQUEST FOR ADDITIONAL INFORMATION

COMMONWEALTH EDISON COMPANY

TO SUPPORT REVIEW OF

GSEP REVISION 93-01

1. During the meeting held in September 1994, the NRC requested ComEd to propose how to resolve the issue of timely activation of the interim EOF. In its letter of November 22, 1994, ComEd committed to demonstrate a "real-time" activation of its emergency plan. This activation would permit the NRC to observe the augmentation of the emergency organization in a realistic fashion which would test as much of the augmentation system as possible without placing undue burden on the licensee. ComEd proposed the demonstration as part of the Braidwood exercise in March 1995 in order to minimize the burden on its staff. During the Braidwood exercise (postponed until April) a new callout system which had been installed to enhance ComEd's augmentation process, failed. A controller had to intervene to inform the exercise participants of the failure and to direct them to use the backup callout systems. As a result of this failure, ComEd was unable to demonstrate its ability to augment the emergency response organization within about 1 hour.

   Two issues were raised as a result of the staff's observation of the Braidwood exercise. They are:

   (a) ComEd has established a Nuclear Duty Officer (NDO) position who activates the callout system. The NDO carries a pager and a cellular phone. The NDO receives a page from the affected plant when an Alert is declared. ComEd indicated that the page may take as long as 15 minutes to reach the NDO. Upon receiving the page from the affected plant, the NDO remotely activates the callout system via a pager and then receives confirmation of the callout system operation by way of a third page. This system of activation is complicated and could result in delaying augmentation of ComEd's emergency response.

   (b) During the Braidwood exercise the NDO was not aware that the callout system had failed. ComEd explained that the callout system (located in the licensee's computer data center in Joilet, IL., which is remote from the NDO) should send the NDO a signal via his pager upon failure of the system. One signal is sent if the system fails to initiate. A second signal is sent if the system fails to complete its callout. The staff is concerned with the time delay in the notification of the NDO of possible system failures. ComEd has stated that a backup system is to be installed which should increase the reliability of the callout system. Notwithstanding this improvement, the staff is still concerned with the lack of positive indication to the NDO that the system is working properly during an emergency callout.

The above concerns have the potential to impact ComEd's capability to augment the Corporate EOF within about one hour from the time of declaration of an Alert and with a high degree of reliability. Please explain how ComEd proposes to resolve this concern.

2. Following the Braidwood exercise, the NRC reviewed augmentation drill results and the procedure for performing it. Review of the drill results indicated that it takes about $1\frac{1}{2}$ hours for ComEd to staff the Corporate EOF. In addition, the following concerns regarding the procedure for conducting off-hour augmentation drills were identified:

   (a) The procedure indicates an acceptance value of $\leq$75 minutes.

   (b) The same procedure indicates that "zero time" is when the NDO receives the notification call. This can raise the actual "acceptable" activation time to 90 minutes.

   In both of the procedure citations above, the activation time would be greater than the goal of one hour from time of declaration of an Alert as stated in Generating Stations' Emergency Plan.

   Please indicate how the above are consistent with the staffing goal stated in ComEd's Generating Stations' Emergency Plan.

Commonwealth Edison Company
1400 Opus Place
Downers Grove, IL 60515-5701

ComEd

May 8, 1996

U. S. Nuclear Regulatory Commission
Washington, DC 20555

Attn.: Document Control Desk

Subject: Braidwood Nuclear Power Station Units 1 and 2
Byron Nuclear Power Station Units 1 and 2
Dresden Nuclear Power Station Units 2 and 3
LaSalle County Nuclear Power Station Units 1 and 2
Quad Cities Nuclear Power Station Units 1 and 2
Zion Nuclear Power Station Units 1 and 2

ComEd Request for Exemption From the Requirements of 10 CFR 50.4(b)(6) For the Distribution of the Updated Final Safety Analysis Report (UFSAR)

NRC Docket Nos. 50-456 and 50-457
NRC Docket Nos. 50-454 and 50-455
NRC Docket Nos. 50-237 and 50-249
NRC Docket Nos. 50-373 and 50-374
NRC Docket Nos. 50-254 and 50-265
NRC Docket Nos. 50-295 and 50-304

Pursuant to 10 CFR 50.12(a), ComEd requests an exemption from the requirements of 10 CFR 50.4(b)(6) regarding the distribution of additional copies of the Updated Final Safety Analysis Reports for each of the six ComEd sites. 10 CFR 50.4(b)(6) specifies the following requirement:

"(6) *Updated FSAR.* An updated Final Safety Analysis Report (FSAR) or replacement pages, pursuant to §50.71(e) must be submitted as follows: the signed original and 10 copies to the Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, one copy to the appropriate Regional Office, and one copy to the appropriate NRC Resident Inspector if one has been assigned to the site of the facility."

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ComEd believes the requirement to submit 10 additional copies to the NRC staff is an unwarranted administrative burden imposed upon licensees without corresponding benefit. It is unclear if the basis for the original requirement is appropriately maintained. Advances in electronic processing of information and document reproduction have rendered the total compliance toward this regulation obsolete. As such, in lieu of submitting ten additional copies, as discussed in 10 CFR 50.4(b)(6), to the Nuclear Regulatory Commission (NRC), ComEd proposes to submit seven total copies to the NRC staff. A more complete discussion regarding the basis for the exemption request is provided as an attachment to this letter.

ComEd believes the cost savings recognized by the aforementioned exemption from the requirements of 10 CFR 50.4(b)(6) is significant. This cost is multiplied by the requirements specified within 10 CFR 50.71(e)(4) for biennial UFSAR revision submittals to the NRC staff. When this cost savings is computed over the life of the stations the savings would be substantial: Byron/Braidwood (15 submittals), Dresden (10 submittals), LaSalle (13 submittals), Quad Cities (10 submittals), and Zion (10 submittals).

I affirm that the content of this transmittal is true and correct to the best of my knowledge information and belief.

In order to realize the cost benefits associated with this exemption request, ComEd requests review and approval by the NRC staff within six months of receipt of this request.

If there are any questions concerning this submittal, please contact this office.

Sincerely,

John C. Brons

John C. Brons
Vice-President

Attachment: Justification for Exemption From the Requirements of 10 CFR 50.4(b)(6)

Subscribed and sworn to before me, a Notary Public in and for the State and County above named, this 8th day of May, 1996. My Commission expires on July 21, 1996.

Betty Fox

Notary Public

"OFFICIAL SEAL"
Betty Fox
Notary Public, State of Illinois
My Commission Expires 7/21/96

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cc: H. Miller, Regional Administrator - RIII
R. Capra, Director of Directorate III-2, NRR
G. Dick, Byron Project Manager - NRR
R. Assa, Braidwood Project Manager - NRR
J. Stang, Dresden Project Manager - NRR
D. Skay, LaSalle Project Manager - NRR
R. Pulsifer, Quad Cities Project Manager - NRR
C. Shiraki, Zion Project Manager - NRR
C. Phillips, Senior Resident Inspector - Braidwood
H. Peterson, Senior Resident Inspector - Byron
C. Vanderniet, Senior Resident Inspector - Dresden
P. Brochman, Senior Resident Inspector - LaSalle
C. Miller, Senior Resident Inspector - Quad Cities
R. Westberg, Senior Resident Inspector - Zion
Office of Nuclear Facility Safety - IDNS

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ATTACHMENT

JUSTIFICATION FOR EXEMPTION FROM THE REQUIREMENTS OF 10 CFR 50.4(b)(6) - NRC STAFF UFSAR DISTRIBUTION

EXEMPTION

Pursuant to 10 CFR 50.12(a), ComEd requests an exemption from the requirements of 10 CFR 50.4(b)(6) regarding the distribution of additional copies of the Updated Final Safety Analysis Reports (UFSAR), including Fire Protection updates, for each of the six ComEd sites. ComEd proposes reducing the total number of copies provided to the NRC staff from thirteen to seven. The seven copies of the UFSAR will be appropriately distributed to the NRC staff offices.

DISCUSSION

10 CFR 50.4(b)(6) specifies the following requirement:

> "(6) *Updated FSAR.* An updated Final Safety Analysis Report (FSAR) or replacement pages, pursuant to ξ50.71(e) must be submitted as follows: the signed original and 10 copies to the Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, one copy to the appropriate Regional Office, and one copy to the appropriate NRC Resident Inspector if one has been assigned to the site of the facility."

ComEd believes the requirement to submit 10 additional copies to the NRC staff is an unwarranted administrative burden imposed upon licensees without corresponding benefit. It is unclear if the basis for the original requirement is appropriately maintained. Advances in electronic processing of information and document reproduction have rendered the total compliance toward this regulation obsolete. As such, in lieu of submitting ten additional copies, as discussed in 10 CFR 50.4(b)(6), to the Nuclear Regulatory Commission (NRC), ComEd proposes to submit seven total copies to the NRC staff. However, to ensure that appropriate NRC staff personnel continue to receive the necessary minimum quantity of UFSAR information, ComEd will submit copies to the following NRC staff locations:

- Original to the Document Control Desk
- Four copies to NRR staff offices (to be used for the NRR Project Manager, the NRR Emergency Response Center, the NRR Operations Area, the NRR Office of General Counsel)
- One copy to the Region III Office
- One copy to the Site Senior Resident Inspector's Office.

The above distribution ensures sufficient updates are distributed to all appropriate NRC staff office locations.

ATTACHMENT (continued)

ComEd's proposed exemption request continues to ensure that the NRC staff receives adequate updated information from the sites regarding the latest version of the Updated Final Safety Analysis Report. In addition, the maintenance of the Document Control Desk on the distribution list ensures that the availability of overall general public information is not adversely hindered.

The proposed exemption request is the reduction of an administrative requirement and has no impact on the safe operation of the facility. As such, the proposed exemption request satisfies the requirements specified by 10 CFR 50.12 as discussed below.

BASIS

A. Criteria for Granting Exemptions Are Met per 10 CFR 50.12(a)(1)

1. *The Requested Exemptions and the Activities Which Would be Allowed Thereunder Are Authorized by Law*

   If the criteria established in 10 CFR 50.12(a) are satisfied, as they are in this case, and if no other prohibition of law exists to preclude the activities which would be authorized by the requested exemption, and there are no such prohibitions, the Commission is authorized by law to grant this exemption request.

2. *The Requested Exemption Will Not Present Undue Risk to the public health and safety.*

   The proposed exemption request is the reduction of an administrative requirement and has no impact on the safe operation of the facility. The maintenance of the Document Control Desk on the distribution list ensures that the availability of overall general public information is not adversely hindered. Therefore, Public participation and knowledge regarding revisions to the plant's UFSAR will not be adversely impacted by the proposed exemption. As such, the requested exemption will not present undue risk to the public health and safety.

3. *The Requested Exemption is consistent with the common defense and security.*

   The proposed exemption request is a purely administrative change that does not affect the operation of the facility in any manner. As such, the common defense and security are unaffected by the proposed exemption request.

B. At Least One of the Special Circumstances Are Present Per 10 CFR 50.12(a)(2)

1. *The Requested Exemptions Will Avoid Undue Hardship or Costs*

The requested exemption is proposed to reduce undue costs associated with the dissemination of redundant information to the NRC staff offices. This cost is multiplied by the requirements specified within 10 CFR 50.71(e)(4) for biennial UFSAR revision submittals to the NRC staff. Based upon the current license expiration dates associated with each of the six ComEd nuclear stations, ComEd projects the cost savings associated with the aforementioned exemption to be significant. As such, the requested exemption will avoid undue costs.

Commonwealth Edison Company
Dresden Generating Station
6500 North Dresden Road
Morris, IL 60450 3600
Tel 815-942-2920

March 14, 1996

ComEd

JSP Ltr: #96 - 0031

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

Subject: Dresden Nuclear Power Station Units 2 and 3
Updated Final Safety Analysis Report, Revision
NRC Docket Numbers 50-237 and 50-249

Reference:

1) B Rybak letter to NRC Document Control Desk, dated December 15, 1995, transmitting UFSAR, Revision 1.

2) J. S. Perry letter to NRC Document Control Desk, dated February 15, 1996, concerning administrative errors in the UFSAR, Revision 1 submittal.

3) P.L. Piet letter to J.B. Martin dated December 30, 1993, transmitting Dresden Rebaselined UFSAR

Reference 1 transmitted the biennial update of the Dresden Station UFSAR (Revision 01, December 1995) to the Document Control Desk in accordance with 10 CFR 50.71. Reference 2 notified you that we had identified administrative errors in the Dresden Station UFSAR, revision 1 submittal.

This letter transmits a corrected biennial update (Revision 01a, December 1995) of the Dresden UFSAR, in accordance with 10 CFR 50.71(e). The changes included in Revision 01a, are changes to the facility and its procedures and are current through June 30, 1995.

This revision accurately represents changes made since the submittal of the rebaselined UFSAR (reference 3), as necessary, to reflect information and analyses or prepared pursuant to Commission requirement and also represents changes made under the provisions of 10 CFR 50.59.

Attached to this letter is a detailed change log which identifies and explains all changes from the rebaselined Dresden UFSAR. This is being provided as an aid to the Staff to assist in the review of the changes made.

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Revision 01a of the Dresden UFSAR supersedes Revision 01 in entirety. Please discard the previous submittal (revision 01), and insert, as directed in the attached instructions, the corrected pages (revision 01a).

Pursuant to 10 CFR 50.4 (b) (6), one (1) signed original and ten (10) copies are being provided to the Document Control Desk, plus one (1) copy to the NRC Region III office and one (1) copy to the Dresden Senior Resident Inspector office.

To the best of my knowledge and belief, the statements contained above are true and correct. In some respect these statements are not based on my personal knowledge, but obtained information furnished by other Commonwealth Edison employees, contractor employees, and consultants. Such information has been reviewed in accordance with Company practice, and I believe it to be reliable.

Please address any questions or comments regarding this submittal to this office.

Very truly yours,

J. Stephen Perry
Vice President
BWR Operations

cc: H. J. Miller, Regional Administrator, NRC, Region III
J. F. Stang, Project Manager, NRR (Unit 2/3)
C. L. Vanderniet, Senior Resident Inspector, Dresden
Office Of Nuclear Facility Safety, IDNS
File: Numerical

bcc: Denny Farrar w/o attachment
Bob Rybak
Chron
Subject File, FSAR
DCDL, electronic partial version
File: SVP Numerical

PAGE CHANGE INDEX / SUMMARY OF CHANGES

PAGE CHANGE INSTRUCTIONS		SUMMARY OF CHANGES
Remove Page (s)	Insert Page (s)	
All existing pages	List of Effective Pages 1 through 23	List of Effective Pages showing Rev. 01A changes.
Contents Page (Revision 0) (For Volume 1)	Contents Page (Rev. 01A/Dec. 1995)	Contents page at front of every volume indicates revision level of set.
Table 1.1-1 Sheet 3 of 3	Table 1.1-1 Sheet 3 of 3	Added Reference of Acronym RVWLIS per addition of the RVWLIS Backfill Modification M12-2(3)-93-004
1.2-14	1.2.14	Updated description of Refueling Operations procedures DMP 0200-13 and DMP 0200-14.
1.2-17	1.2-17	Updated text to incorporate Gaseous Monitoring System and Fuel Storage Building Ventilation Modification M12-0-91-007
1.2-19	1.2-19	Updated paragraph to include revised text to incorporate Replacement of Valve 2-4608 PCV-4601 per P12-2-94-265
Figure 1.2-2 Rev. F Figure 1.2-3 Rev. E Figure 1.2-4 Rev. K Figure 1.2-9 Rev. A Figure 1.2-13 Rev. A	Figure 1.2-2 Rev. H Figure 1.2-3 Rev. G Figure 1.2-4 Rev. L Figure 1.2-9 Rev. B Figure 1.2-13 Rev. B	Insert Latest Revision.
Figure 1.7-1 Rev. G Figure 1.7-2 Rev. Q	Figure 1.7-1 Rev. H Figure 1.7-2 Rev. R	Insert the latest drawing revisions.
2.4-1 2.4-3 Table 2.4-1 Sheet 1 of 1	2.4-1 2.4-3 Table 2.4-1 Sheet 1 of 1	Correct text to show maximum historical flood elevation. Remove reference to nominal flood level. Clarify flood values, high river values, and remove incorrect max flood values in Table.
3.1-1	3.1-1	Revised paragraph per replacement of valve 2-4608-PCV-4601 in the Unit 2 Air System P12-2-94-265.
3.3-8	3.3-8	Revised paragraph to incorporate the Isolation Condenser Upgrade Modification M12-2-90-057 changes.
3.7-3	3.7-3	Updated paragraph to incorporate Refueling Platform Replacement Exempt Plant Changes P12-2-93-280 and P12-3-93-273.
N/A	3.8-7-a	Addition of text to incorporate RVWLIS Backfill Modification M12-2(3)-93-004 changes.

PAGE CHANGE INDEX / SUMMARY OF CHANGES

PAGE CHANGE INSTRUCTIONS		SUMMARY OF CHANGES
Remove Page (s)	Insert Page (s)	
Table 3.8-2 Sheet 1 of 1	Table 3.8-2 Sheet 1 of 1	Revised table to incorporate new penetrations for 3 separate Minor Plant Changes. P12-3-92-714: Split Penetration X-111A into A & B and 111A changed to a type 1A; P12-3-92-715: Changed penetration X-138 to type 1A and P12-3-92-716 split penetration X-149 into A & B and changed B to a type 1A.
Table 3.8-4 Sheet 3 of 3	Table 3.8-4 Sheet 3 of 3	Updated table to incorporate the RVWLIS Backfill Modification Changes to Note #2 to show reference to Unit 2 & Unit 3 and to correct RVWLIS acronym.
Contents Page (Revision 0) (For Volume 2)	Contents Page (Rev. 01A/Dec. 1995)	Contents page at front of every volume indicates revision level of set.
4.4-6 Figure 4.4-1	4.4-6 Figure 4.4-1	Revised paragraph and figure to show 70% Flow Control Line per Commitment to Dresden Response to GL 94-02.
4.6-8 4.6-9	4.6-8 4.6-9	Added Reference E to Section 4.6.3.3.1 and updated 2nd paragraph in Section 4.6.3.3.2 to incorporate RVWLIS Backfill Modification M12-2(3)-93-004.
Figure 4.6-4 Rev. BB Figure 4.6-4 Rev. AAS	Figure 4.6-4 Rev. BE Figure 4.6-5 Rev. AAU	Insert latest revisions.
5.2-2	5.2-2 5.2-2-a	Added paragraph to Section 5.2.2.2 that references NUREG 0737, Item II.D.1 Additional Evaluation of Relief & Safety Valve Testing.
5.2-14	5.2-14	Updated paragraph to incorporate RWCU Pipe Replacement Schedular Commitment to the NRC.
5.2-22	5.2-22	Delete section 5.2.5.6.2 per Plant Design Change M12-3-92-001C.
Contents Page (Revision 0) (For Volume 3)	Contents Page (Rev. 01A/Dec. 1995)	Contents page at front of every volume indicates revision level of set.
5.4-20	5.4-20	Editorial correction

PAGE CHANGE INDEX / SUMMARY OF CHANGES

PAGE CHANGE INSTRUCTIONS		SUMMARY OF CHANGES
Remove Page (s)	Insert Page (s)	
5.4-27 5.4-28 5.4-29 5.4-31 5.4-32 5.4-36	5.4-27 5.4-28 5.4-29 5.4-31 5.4-32 5.4-36	Revised Sections 5.4.6.2 and 5.4.6.3 to incorporate Isolation Condenser Make-Up Pump Upgrade Modification M12-2(3)-90-057 Partials. Change paragraph 5.4.7.2 to correct configuration of shutdown cooling injection path to vessel. (page 5.4-31) Added safety related references to section 5.4.6.2 and 5.4.7.2 to differentiate from the non-safety related 250 VDC Battery System Modification M12-2(3)-92-005a addition. Changed RWCU high pressure alarm set from 150 psig to 130 psig per set point change SPC #3-95-022 (p5.4-36).
Figure 5.4-1 Rev. AK Figure 5.4-2 Rev. HQ Figure 5.4-3 Rev. AV Figure 5.4-4 Rev. AJ Figure 5.4-15 Rev. C Figure 5.4-18 Rev. KR Figure 5.4-19 Rev. AM Figure 5.4-21 Rev. AF Figure 5.4-23 Rev. ZK Figure 5.4-24 Rev. AS Figure 5.4-26 Rev. Z Figure 5.4-27 Rev. S	Figure 5.4-1 Rev. AS Figure 5.4-2 Rev. HT Figure 5.4-3 Rev. AZ Figure 5.4-4 Rev. AM Figure 5.4-15 Rev. D Figure 5.4-18 Rev. KW Figure 5.4-19 Rev. AS Figure 5.4-21 Rev. AH Figure 5.4-23 Rev. ZL Figure 5.4-24 Rev. AV Figure 5.4-26 Rev. AA Figure 5.4-27 Rev. T	Insert the latest drawing revisions.

PAGE CHANGE INDEX / SUMMARY OF CHANGES

PAGE CHANGE INSTRUCTIONS		SUMMARY OF CHANGES
Remove Page (s)	Insert Page (s)	
6.2-4 6.2-14 6.2-17 6.2-70 6.2-74 6.2-79 6.2-80 6.2-81 6.2-88 6.2-95	6.2-4 6.2-14 6.2-17 6.2-70 6.2-74 6.2-74-a 6.2-79 6.2-80 6.2-81 6.2-88 6.2-95	Added RBVS reference to Containment Venting Section and revised the Vent, Purge, and Inerting System Section to show venting the containment during normal operation is allowed (pages 6.2-14, 6.2-80, 6.2-81) Typographical error correction in suppression chamber sizing determination - psia to psig (page 6.2-17) Revised instrument line excess flow check valves sentence to read "and simple" check valves, per RVWLIS Backfill Modification M12-2(3)-93-004 (page 6.2-70) Updated Section 6.2.4.3.2 Containment Integrity to incorporate Minor Plant Changes P12-2-93-220 and P12-3-93-226 upgrade information (pages 6.2-74, 6.2-74a) Clarification of ACAD/NCAD description to satisfy a Corrective Action specified in LER 2-95-011 (pages 6.2-79, 6.2-88) Revised Personnel Airlock Door and Personnel Access Lock Double Door description per: SER/TER info that forwards exemption from certain 10CFR 50.54 (O) & APP.O Requirements (pages 6.2-4, 6.2-95)
Table 6.2-9 Sheets 1 through 10	Table 6.2-9 Sheets 1 through 10 (Size 11 x 17)	Multiple changes include: (Sheet 1 of 10) changed penetration X-108A valves 1301-1 and 1301-2 max. iso. times from 30 to 40 seconds. Per: M12-2-92-001 partials C & D. Penetration X-108A valves 1301-17 and 1301-20 changed max. iso. times from 5 to 10 seconds. Per: DATR 3/4.18 and added note to X-106 valves 220-1 and 22-2 showing Unit 3 valves were changed from gate to globe valves. (Sheet 2 of 10) changed penetration X-109B(A) valves 1301-3 and 1301-4 max. iso. times from 30 to 40 seconds and X-115A(128) valves 2301-4 and 2301-5 max. iso. times from 25 to 50 seconds per M12-2-92-001 partials C, D, E, F & G. Also added note to X-113 valve 1201-1A showing Unit 3 valve was changed from a globe to a gate valve. (Sheet 5 of 10) changed penetration X-147 valve 205-24 max. iso. times from 15 to 45 seconds per: P12-3-93-279
Table 6.2-10 Sheets 1 through 3	Table 6.2-10 Sheets 1 through 3	Changed Drywell Equipment Drains Penetration X-118 valve number from 3-2099-553 to 3-2099-552 per design changes P12-2-93-220 and P12-3-93-226.

PAGE CHANGE INDEX / SUMMARY OF CHANGES

PAGE CHANGE INSTRUCTIONS		SUMMARY OF CHANGES
Remove Page (s)	Insert Page (s)	
Table 6.2-11 Sheets 1 through 3	Table 6.2-11 Sheets 1 through 3	Changed Drywell Equipment Drains Penetration X-118 valve number from 3-2099-553 to 3-2099-552 per design changes P12-2-93-220 and P12-3-93-226.
Figure 6.2-12 Rev. BT Figure 6.2-13 Rev. BB	Figure 6.2-12 Rev. CC Figure 6.2-13 Rev. BE	Insert the latest drawing revisions.
Contents Page (Revision 0) (For Volume 4)	Contents Page (Rev. 01A/Dec. 1995)	Contents page at front of every volume indicates revision level of set.
6.3-18 6.3-73	6.3-18 6.3-73	Added "Safety Related" to differentiate between the 250 Vdc Non-Safety Related Battery System per: M12-2(3)-92-005A
Figure 6.3-2A Rev. YE Figure 6.3-2B Rev. BC Figure 6.3-7A Rev. AX Figure 6.3-7B Rev. UM Figure 6.3-9A Rev. AS Figure 6.3-9B Rev. BB	Figure 6.3-2A Rev. YL Figure 6.3-2B Rev. BG Figure 6.3-7A Rev. BB Figure 6.3-7B Rev. UR Figure 6.3-9A Rev. AX Figure 6.3-9B Rev. BH	Insert the latest drawing revisions.
6.4-4 6.4-5	6.4-4 6.4-5	Clarified description of the 2/3 Control Room Ventilation System during normal and emergency pressurized modes of operation.
Table 6.5-1 Sheet 1 of 1	Table 6.5-1 Sheet 1 of 1	Changed units to inches of H_20 rather than feet of H_20 it was previously labeled incorrectly.. Changed per: FSAR Q & A B.10
Figure 6.5-1 Rev. PU	Figure 6.5-1 Rev. PY	Insert the latest drawing revision.
Figure 6.5-2	Figure 6.5-2	SBGT carbon adsorber trays were replaced in design change P12-0-91-694. Figure reflects new tray.
Figure 6.5-3	Figure 6.5-3	Changed units to inches of H_20 for Standby Gas Treatment System Exhaust Fan Static Pressure. It was labeled incorrectly as psid.
Table 7.2-1 Sheet 1 of 1	Table 7.2-1 Sheet 1 of 1	Changed Condenser Low Vacuum Scram Setpoint from 23 in. to 21 in. per Tech Spec Amendments #134 and #128 for Units 2 & 3 respectively.
Table 7.3-1 Sheet 1 of 1	Table 7.3-1 Sheet 1 of 1	Changed HPCI Steam Line High Flow from "150 in H_20 differential to "Less than or Equal to 300% Rated Steam Flow"
7.6-20 7.6-21 7.6-22	7.6-20 7.6-21 7.6-22 7.6-22-a	Added text to Section 7.6.2.2.1 to incorporate minor plant change P12-2-91-698 "Reactor Vessel Shell and Flange Thermocouple Replacement" and Section 7.6.2.2.3 to incorporate Modification M12-2(3)-93-004 "RVWLIS Backfill" changes.

PAGE CHANGE INDEX / SUMMARY OF CHANGES

PAGE CHANGE INSTRUCTIONS		SUMMARY OF CHANGES
Remove Page (s)	Insert Page (s)	
Contents Page (Revision 0) (For Volume 5)	Contents Page (Rev. 01A/Dec. 1995)	Contents Page at front of every volume indicates revision level of set.
8.1-2	8.1-2	Revised paragraph in Section 8.1.3 to differentiate between the Safety Related and Non-Safety Related 250 Vdc Power System per: M12-2(3)-92-005A
8.3-4	8.3-4	Update Section 8.3.1.2 4160V System to include 27N-R Relays per the second level undervoltage relay replacement for Units 2 & 3.
8.3-5	8.3-5	Updated Section 8.3.1.2.1 System Description to include reference to Buses 23-1 and 33-1 second manual crosstie connection between Unit 2 and Unit 3 per M12-0-91-018 partials A & B.
8.3-8 8.3-9 8.3-19 8.3-20 8.3-21 8.3-24 8.3-25 8.3-26 8.3-27	8.3-8 8.3-9 8.3-19 8.3-20 8.3-21 8.3-21-a 8.3-24 8.3-25 8.3-26 8.3-27	Updated Section 8.3 in multiple areas to include reference to the Non-Safety Related 250 Vdc Battery System per M12-2(3)-92-005A.
Table 8.3-1 Sheet 1 of 7 Table 8.3-1 Sheet 4 of 7 Table 8.3-1 Sheet 7 of 7	Table 8.3-1 Sheet 1 of 7 Table 8.3-1 Sheet 4 of 7 Table 8.3-1 Sheet 7 of 7	Changed 4160 kV Buses 23-1 and 24-1 Bus Tie from 3 to 4 - AMH 4.16 - 250 - MVA, 1200 A per M12-0-91-018 A & B on Sheet 1 of 7. Added "Safety Related" to the 2-250-V Battery Chargers reference on Sheet 4 of 7 per M12-2(3)-92-005A, and added notes to Sheet 7 of 7 reflecting the upgrade of Bus 33 from a 250 MVA to a 350 MVA rating per M12-0-91-019F.
Table 8.3-8 Sheet 1 of 1	Table 8.3-8 Sheet 1 of 1	Added "Safety-Related" to 250 Vdc System Table Reference per M12-2(3)-92-005A
Figure 8.3-1 Rev. G Figure 8.3-2 Rev. AG Figure 8.3-9 Rev. C Figure 8.3-10 Rev. B	Figure 8.3-1 Rev. L Figure 8.3-2 Rev. AH Figure 8.3-9 Rev. D Figure 8.3-10 Rev. C	Insert the latest drawing revisions.
9-i 9-ii 9-iii	9-i 9-ii 9-iii	Revise Table of Contents.
9-vi 9-vii	9-vi 9-vii	Revise list of figures

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9.1-3 9.1-5 9.1-11 9.1-15 9.1-16 9.1-17 9.1-21 9.1-22	9.1-3 9.1-5 9.1-5-a 9.1-11 9.1-11-a 9.1-15 9.1-16 9.1-17 9.1-21 9.1-22	Sections 9.1.2.2.2 and 9.1.4.2.3 updated to include Reactor Steam Dryer Removal and Installation changes to procedures DMP 0200-13 and DMP 0200-14 (pages 9.1-3, 9.1-21) Revised Section 9.1.2.2.3.2 Title and added new Section 9.1.2.2.3.3 Spent Fuel Pool Blade Guide Racks and 9.1.2.3.3 Spent Fuel Blade Guide Racks per M12-2(3)-84-120. (pages 9.1-5, 9.1-5a, 9.1-11) Revised pages to include the Refueling Platform Replacement changes per P12-2-93-280 and P12-3-93-273 (pages 9.1-15, 9.1-16, 9.1-17, 9.1-22)
Table 9.1-1 Sheet 1 of 1	Table 9.1-1 Sheet 1 of 1	Updated storage equipment references to include Spent Fuel Pool High Density Racks and Spent Fuel Pool Blade Guide Storage Racks per M12-2(3)-84-120 and added Pole Handling System and Mast Mounted Camera reference to Servicing Aids per P12-2-93-280 and P12-3-93-273. (Delete obsolete Figure 9.1-15)
Figure 9.1-3 Rev. AR Figure 9.1-4 Rev. AD Figure 9.1-13 Rev. AD Figure 9.1-14 Rev. V	Figure 9.1-3 Rev. AT Figure 9.1-4 Rev. AG Figure 9.1-13 Rev. AE Figure 9.1-14 Rev. X	Insert the latest drawing revisions.
Figure 9.1-15	N/A	Figure 9.1-15 Deleted.
9.2-19	9.2-19	Updated Section 9.2.6.3 Safety Evaluation to include Unit 2 & 3 480 Vdc power reference per M12-2(3)-90-057 partials.
Figure 9.2-1 Rev. S Figure 9.2-2 Rev. P Figure 9.2-3 Rev. BN Figure 9.2-4 Rev. NV Figure 9.2-8 Rev. KQ Figure 9.2-9 Rev. AG Figure 9.2-10 Rev. CB Figure 9.2-11 Rev. A Figure 9.2-12 Rev. A Figure 9.2-15 Rev. A Figure 9.2-16 Rev. A Figure 9.2-19 Rev. AJ Figure 9.2-20 Rev. LU Figure 9.2-21 Rev. AD Figure 9.2-22 Rev. AH	Figure 9.2-1 Rev. Y Figure 9.2-2 Rev. W Figure 9.2-3 Rev. BY Figure 9.2-4 Rev. PC Figure 9.2-8 Rev. KS Figure 9.2-9 Rev. AH Figure 9.2-10 Rev. CH Figure 9.2-11 Rev. B Figure 9.2-12 Rev. B Figure 9.2-15 Rev. C Figure 9.2-16 Rev. B Figure 9.2-19 Rev. AN Figure 9.2-20 Rev. LX Figure 9.2-21 Rev. AF Figure 9.2-22 Rev. AL	Insert the latest drawing revisions.

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9.3-1 9.3-2 9.3-3 9.3-4 9.3-5 9.3-6 9.3-7 9.3-8 9.3-9 9.3-10 9.3-11 9.3-12 9.3-13 9.3-14 9.3-15 9.3-16 9.3-17 9.3-18 9.3-19 9.3-20 9.3-21 9.3-22 9.3-23 9.3-24 9.3-25 9.3-26 9.3-27 9.3-28 9.3-29 9.3-30	9.3-1 9.3-2 9.3-3 9.3-4 9.3-4-a 9.3-5 9.3-6 9.3-7 9.3-8 9.3-9 9.3-10 9.3-11 9.3-12 9.3-13 9.3-14 9.3-15 9.3-16 9.3-17 9.3-18 9.3-19 9.3-20 9.3-21 9.3-22 9.3-23 9.3-24 9.3-25 9.3-26 9.3-27 9.3-28 9.3-29 9.3-30 9.3-31 9.3-32 9.3-33 9.3-34 9.3-35 9.3-36	Multiple changes to Section 9.3 including total re-write of 9.3.2.1 High Radiation Sampling System to reflect the Post TMI NUREG 0737 & R.G 1.97 requirements. Revised text reflects as built differences from preliminary design. Submitted per NRC Commitment Reduction Plan, other changes include incorporation of P12-2-94-265 changes to 9.3.5.2 and 9.3.5.4, and P12-2-93-205 changes to 9.3.1.3.1.
Figure 9.3-3 Rev. AN Figure 9.3-4 Rev. E	Figure 9.3-3 Rev. AQ Figure 9.3-4 Rev. G	Insert the latest drawing revisions.
Figure 9.3-5	Figure 9.3-5	Figure updated per P12-2-93-205 changes and P12-2-94-265 changes.
Figure 9.3-8 Rev. HM	Figure 9.3-8 Rev. HQ	Insert the latest drawing revisions.
9.4-6 9.4-7	9.4-6 9.4-7	Editorial Change

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Figure 9.4-1 Rev. B Figure 9.4-2 Rev. 0 Figure 9.4-3 Rev. 7 Figure 9.4-4 Rev. AA Figure 9.4-7 Rev. D Figure 9.4-8 Rev. E Figure 9.4-9 Rev. M Figure 9.4-10 Rev. G Figure 9.4-11 Rev. F Figure 9.4-12 Rev. E Figure 9.4-13 Rev. D Figure 9.4-14 Rev. D Figure 9.4-15 Rev. M Figure 9.4-16 Rev. L Figure 9.4-17 Rev. E	Figure 9.4-1 Rev. C Figure 9.4-2 Rev. E Figure 9.4-3 Rev. 10 Figure 9.4-4 Rev. AC Figure 9.4-7 Rev. E Figure 9.4-8 Rev. F Figure 9.4-9 Rev. P Figure 9.4-10A Rev. B Figure 9.4-10B Rev. A Figure 9.4-11 Rev. G Figure 9.4-12 Rev. F Figure 9.4-13 Rev. E Figure 9.4-14 Rev. E Figure 9.4-15 Rev. Q Figure 9.4-16 Rev. Q Figure 9.4-17 Rev. F	Insert the latest drawing revisions.
9.5.1	9.5.1	Added paragraph to Section 9.5.2.2 to incorporate P12-0-93-201 changes on Cellular Phone Antenna Installation
9.5-3 9.5-4	9.5-3 9.5-3-a 9.5-4	Re-write of Section on Intraplant Radio Communication to incorporate Minor Plant Change P12-0-92-603 Completion of 900 MHz Radio Installation.
9.5-8	9.5-8	Revised DGCW Flow Requirements per Ler 2-93-018.
Figure 9.5-1 Rev. M	Figure 9.5-1 Rev. Q	Insert the latest drawing revision.
Contents Page (Revision 0) (For Volume 6)	Contents Page (Rev. 01A/Dec. 1995)	Contents page at front of every volume indicates revision level of set.
10.2-2	10.2-2 10.2-2-a	Deleted old paragraph on FAS Subsystem and replaced with new re-writes for Unit 2 and Unit 3 per P12-3-93-249, D-3 EHC Tubing Upgrade.
Figure 10.3-1 Rev. NP Figure 10.3-2 Rev. AAE Figure 10.3-3 Rev. AG Figure 10.3-4 Rev. NV	Figure 10.3-1 Rev. NS Figure 10.3-2 Rev. AAH Figure 10.3-3 Rev. AJ Figure 10.3-4 Rev. NX	Insert the latest drawing revisions.
10.4-3	10.4-3	Updated Section 10.4.1.5 to incorporate Tech. Spec. Amendments 134 Unit 2 and 138 Unit 3 Condenser Low Vacuum Scram Setpoint change.
Figure 10.4-1 Rev. JY Figure 10.4-7 Rev. X Figure 10.4-8 Rev. KT Figure 10.4-9 Rev. AW Figure 10.4-10 Rev. AA	Figure 10.4-1 Rev. JY Figure 10.4-7 Rev. AB Figure 10.4-8 Rev. KV Figure 10.4-9 Rev. AX Figure 10.4-10 Rev. AB	Insert the latest drawing revisions.

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Remove Page (s)	Insert Page (s)	
	Verify: 11-i (original T.O.C.)	Verify 11-i, T.O.C. is original. If other than original, i.e., Rev. 01 / Dec. 1995, replace with the copy provided.
11.1-3 11.1-10	11.1-3 11.1-10	Updated text to reflect the existing noncontaminated drains within the RCA and associated administrative controls per LER 237-93-022.
11.2-18 11.2-20	11.2-18 11.2-20	Updated text to reflect the existing noncontaminated drains within the RCA and associated administrative controls per LER 237-93-022.
Figure 11.2-1 Rev. BN Figure 11.2-2 Original Figure 11.2-3 Rev. YF Figure 11.2-4 Rev. MV Figure 11.2-5 Rev. Y Figure 11.2-7 Rev. V Figure 11.2-8 Rev. V Figure 11.2-10 Rev. AJ Figure 11.2-11 Rev. AC Figure 11.2-12 Rev. W Figure 11.2-13 Rev. S Figure 11.2-14 Rev. T Figure 11.2-16 Rev. C Figure 11.2-19 Rev. G Figure 11.2-20 Rev. H	Figure 11.2-1 Rev. BU Figure 11.2-2 Rev. UW Figure 11.2-3 Rev. YJ Figure 11.2-4 Rev. MX Figure 11.2-5 Rev. Z Figure 11.2-7 Rev. W Figure 11.2-8 Rev. X Figure 11.2-10 Rev. AM Figure 11.2-11 Rev. AE Figure 11.2-12 Rev. X Figure 11.2-13 Rev. V Figure 11.2-14 Rev. W Figure 11.2-16 Rev. E Figure 11.2-19 Rev. K Figure 11.2-20 Rev. K	Insert the latest drawing revisions.
Figure 11.2-21	Figure 11.2-21	Revised figure to incorporate Rad Waste Modification M12-2/3-87-002 and changed the Waste Surge Tank to a River Discharge Tank.
Figure 11.3-1 Rev. BL Figure 11.3-4 Rev. HL Figure 11.3-6 Rev. W Figure 11.3-7 Rev. AAE Figure 11.3-8 Rev. NV Figure 11.3-12 Rev. 5 Figure 11.3-13 Rev. AG Figure 11.3-15 Rev. V Figure 11.3-18 Rev. C Figure 11.3-19 Rev. K Figure 11.4-1 Rev. AG	Figure 11.3-1 Rev. BM Figure 11.3-4 Rev. HN Figure 11.3-6 Rev. X Figure 11.3-7 Rev. AAH Figure 11.3-8 Rev. NX Figure 11.3-12 Rev. 6 Figure 11.3-13 Rev. AJ Figure 11.3-15 Rev. X Figure 11.3-18 Rev. D Figure 11.3-19 Rev. M Figure 11.4-1 Rev. AJ	Insert the latest drawing revisions.
11.5-9 11.5-10	11.5-9 11.5-10	Updated Section 11.5.2.3.1 Sping Monitoring Instrumentation to accurately reflect the current methods used for monitoring per DRS 2000-03.
11.5-14	11.5-14 11.5-14-a	Updated Section 11.5.2.7 to reflect the Service Water Radiation Monitoring System Upgrade for Unit 2 P12-2-94-218.

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Table 11.5-1 Sheet 4 of 4	Table 11.5-1 Sheet 4 of 4	Note added to Table Equipment Alarm Types
12.1-1 12.1-2	12.1-1 12.1-2	Eliminated reference to the (CAC) Corporate ALARA Committee per DAP 12-07.
Contents Page (Revision 0) (For Volume 7)	Contents Page (Rev. 01A/Dec. 1995)	Contents Page at front of every volume indicates revision level of set.
13-i 13-ii 13-iii 13-iv 13-v	13-i 13-ii 13-iii 13-iv	Revise T.O.C., List of tables and figures.
13.1-1 13.1-2 13.1-3 13.1-4 13.1-5 13.1-6 13.1-7 13.1-8 13.1-9 13.1-10 13.1-11 13.1-12 13.1-13	13.1-1 13.1-2	Removed corporate and station specific functions and responsibilities for personnel from section 13.1.
Table 13.1-1 Sheet 1 Figure 13.1-1 Figure 13.1-2	N/A	Removed corporate and station specific functions and responsibilities from section 13.1.
13.2-2	13.2-2	Updated paragraph in Section 13.2.1.1.1 to incorporate procedure changes to DRP 5000-05 and DAP 12-35.
13.3-1 13.3-2 13.3-3 13.3-4 13.3-5 13.3-6 13.3-7 13.3-8 13.3-9 13.3-10	13.3-1	Deletion of details already in the G.S.E.P. removal of redundant information.
13.5-8	13.5-8	Revised paragraph in Section 13.5.2.2.15 to incorporate changes to DAP 9-9 Rev. 8 Special Procedures.

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Remove Page (s)	Insert Page (s)	
15.2-10	15.2-10	Changed Item C in Section 15.2.5.2 to scram at 21 in. Hg Vacuum per upgraded Tech Spec amendments DPR 19 Amend. 134 and DRP 25 Amen. 128.
15.4-14	15.4-14	Change in main condenser vacuum scram/trip setpoint from 23 in Hg to 21 in. Hg per upgraded Tech Spec Amendments DPR 19 Amend. 134 and DPR 25 Amend. 128.

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All existing pages	List of Effective Pages 1 through 23	List of Effective Pages showing Rev. 01A changes.
Contents Page (Revision 0) (For Volume 1)	Contents Page (Rev. 01A/Dec. 1995)	Contents page at front of every volume indicates revision level of set.
Table 1.1-1 Sheet 3 of 3	Table 1.1-1 Sheet 3 of 3	Added Reference of Acronym RVWLIS per addition of the RVWLIS Backfill Modification M12-2(3)-93-004
1.2-14	1.2.14	Updated description of Refueling Operations procedures DMP 0200-13 and DMP 0200-14.
1.2-17	1.2-17	Updated text to incorporate Gaseous Monitoring System and Fuel Storage Building Ventilation Modification M12-0-91-007
1.2-19	1.2-19	Updated paragraph to include revised text to incorporate Replacement of Valve 2-4608 PCV-4601 per P12-2-94-265
Figure 1.2-2 Rev. F Figure 1.2-3 Rev. E Figure 1.2-4 Rev. K Figure 1.2-9 Rev. A Figure 1.2-13 Rev. A	Figure 1.2-2 Rev. H Figure 1.2-3 Rev. G Figure 1.2-4 Rev. L Figure 1.2-9 Rev. B Figure 1.2-13 Rev. B	Insert Latest Revision.
Figure 1.7-1 Rev. G Figure 1.7-2 Rev. Q	Figure 1.7-1 Rev. H Figure 1.7-2 Rev. R	Insert the latest drawing revisions.
2.4-1 2.4-3 Table 2.4-1 Sheet 1 of 1	2.4-1 2.4-3 Table 2.4-1 Sheet 1 of 1	Correct text to show maximum historical flood elevation. Remove reference to nominal flood level. Clarify flood values, high river values, and remove incorrect max flood values in Table.
3.1-1	3.1-1	Revised paragraph per replacement of valve 2-4608-PCV-4601 in the Unit 2 Air System P12-2-94-265.
3.3-8	3.3-8	Revised paragraph to incorporate the Isolation Condenser Upgrade Modification M12-2-90-057 changes.
3.7-3	3.7-3	Updated paragraph to incorporate Refueling Platform Replacement Exempt Plant Changes P12-2-93-280 and P12-3-93-273.
N/A	3.8-7-a	Addition of text to incorporate RVWLIS Backfill Modification M12-2(3)-93-004 changes.

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Table 3.8-2 Sheet 1 of 1	Table 3.8-2 Sheet 1 of 1	Revised table to incorporate new penetrations for 3 separate Minor Plant Changes. P12-3-92-714: Split Penetration X-111A into A & B and 111A changed to a type 1A; P12-3-92-715: Changed penetration X-138 to type 1A and P12-3-92-716 split penetration X-149 into A & B and changed B to a type 1A.
Table 3.8-4 Sheet 3 of 3	Table 3.8-4 Sheet 3 of 3	Updated table to incorporate the RVWLIS Backfill Modification Changes to Note #2 to show reference to Unit 2 & Unit 3 and to correct RVWLIS acronym.
Contents Page (Revision 0) (For Volume 2)	Contents Page (Rev. 01A/Dec. 1995)	Contents page at front of every volume indicates revision level of set.
4.4-6 Figure 4.4-1	4.4-6 Figure 4.4-1	Revised paragraph and figure to show 70% Flow Control Line per Commitment to Dresden Response to GL 94-02.
4.6-8 4.6-9	4.6-8 4.6-9	Added Reference E to Section 4.6.3.3.1 and updated 2nd paragraph in Section 4.6.3.3.2 to incorporate RVWLIS Backfill Modification M12-2(3)-93-004.
Figure 4.6-4 Rev. BB Figure 4.6-4 Rev. AAS	Figure 4.6-4 Rev. BE Figure 4.6-5 Rev. AAU	Insert latest revisions.
5.2-2	5.2-2 5.2-2-a	Added paragraph to Section 5.2.2.2 that references NUREG 0737, Item II.D.1 Additional Evaluation of Relief & Safety Valve Testing.
5.2-14	5.2-14	Updated paragraph to incorporate RWCU Pipe Replacement Schedular Commitment to the NRC.
5.2-22	5.2-22	Delete section 5.2.5.6.2 per Plant Design Change M12-3-92-001C.
Contents Page (Revision 0) (For Volume 3)	Contents Page (Rev. 01A/Dec. 1995)	Contents page at front of every volume indicates revision level of set.
5.4-20	5.4-20	Editorial correction

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5.4-27 5.4-28 5.4-29 5.4-31 5.4-32 5.4-36	5.4-27 5.4-28 5.4-29 5.4-31 5.4-32 5.4-36	Revised Sections 5.4.6.2 and 5.4.6.3 to incorporate Isolation Condenser Make-Up Pump Upgrade Modification M12-2(3)-90-057 Partials. Change paragraph 5.4.7.2 to correct configuration of shutdown cooling injection path to vessel. (page 5.4-31) Added safety related references to section 5.4.6.2 and 5.4.7.2 to differentiate from the non-safety related 250 VDC Battery System Modification M12-2(3)-92-005a addition. Changed RWCU high pressure alarm set from 150 psig to 130 psig per set point change SPC #3-95-022 (p5.4-36).
Figure 5.4-1 Rev. AK Figure 5.4-2 Rev. HQ Figure 5.4-3 Rev. AV Figure 5.4-4 Rev. AJ Figure 5.4-15 Rev. C Figure 5.4-18 Rev. KR Figure 5.4-19 Rev. AM Figure 5.4-21 Rev. AF Figure 5.4-23 Rev. ZK Figure 5.4-24 Rev. AS Figure 5.4-26 Rev. Z Figure 5.4-27 Rev. S	Figure 5.4-1 Rev. AS Figure 5.4-2 Rev. HT Figure 5.4-3 Rev. AZ Figure 5.4-4 Rev. AM Figure 5.4-15 Rev. D Figure 5.4-18 Rev. KW Figure 5.4-19 Rev. AS Figure 5.4-21 Rev. AH Figure 5.4-23 Rev. ZL Figure 5.4-24 Rev. AV Figure 5.4-26 Rev. AA Figure 5.4-27 Rev. T	Insert the latest drawing revisions.

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6.2-4 6.2-14 6.2-17 6.2-70 6.2-74 6.2-79 6.2-80 6.2-81 6.2-88 6.2-95	6.2-4 6.2-14 6.2-17 6.2-70 6.2-74 6.2-74-a 6.2-79 6.2-80 6.2-81 6.2-88 6.2-95	Added RBVS reference to Containment Venting Section and revised the Vent, Purge, and Inerting System Section to show venting the containment during normal operation is allowed (pages 6.2-14, 6.2-80, 6.2-81) Typographical error correction in suppression chamber sizing determination - psia to psig (page 6.2-17) Revised instrument line excess flow check valves sentence to read "and simple" check valves, per RVWLIS Backfill Modification M12-2(3)-93-004 (page 6.2-70) Updated Section 6.2.4.3.2 Containment Integrity to incorporate Minor Plant Changes P12-2-93-220 and P12-3-93-226 upgrade information (pages 6.2-74, 6.2-74a) Clarification of ACAD/NCAD description to satisfy a Corrective Action specified in LER 2-95-011 (pages 6.2-79, 6.2-88) Revised Personnel Airlock Door and Personnel Access Lock Double Door description per: SER/TER info that forwards exemption from certain 10CFR 50.54 (O) & APP.O Requirements (pages 6.2-4, 6.2-95)
Table 6.2-9 Sheets 1 through 10	Table 6.2-9 Sheets 1 through 10 (Size 11 x 17)	Multiple changes include: (Sheet 1 of 10) changed penetration X-108A valves 1301-1 and 1301-2 max. iso. times from 30 to 40 seconds. Per: M12-2-92-001 partials C & D. Penetration X-108A valves 1301-17 and 1301-20 changed max. iso. times from 5 to 10 seconds. Per: DATR 3/4.18 and added note to X-106 valves 220-1 and 22-2 showing Unit 3 valves were changed from gate to globe valves. (Sheet 2 of 10) changed penetration X-109B(A) valves 1301-3 and 1301-4 max. iso. times from 30 to 40 seconds and X-115A(128) valves 2301-4 and 2301-5 max. iso. times from 25 to 50 seconds per M12-2-92-001 partials C, D, E, F & G. Also added note to X-113 valve 1201-1A showing Unit 3 valve was changed from a globe to a gate valve. (Sheet 5 of 10) changed penetration X-147 valve 205-24 max. iso. times from 15 to 45 seconds per: P12-3-93-279
Table 6.2-10 Sheets 1 through 3	Table 6.2-10 Sheets 1 through 3	Changed Drywell Equipment Drains Penetration X-118 valve number from 3-2099-553 to 3-2099-552 per design changes P12-2-93-220 and P12-3-93-226.

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Table 6.2-11 Sheets 1 through 3	Table 6.2-11 Sheets 1 through 3	Changed Drywell Equipment Drains Penetration X-118 valve number from 3-2099-553 to 3-2099-552 per design changes P12-2-93-220 and P12-3-93-226.
Figure 6.2-12 Rev. BT Figure 6.2-13 Rev. BB	Figure 6.2-12 Rev. CC Figure 6.2-13 Rev. BE	Insert the latest drawing revisions.
Contents Page (Revision 0) (For Volume 4)	Contents Page (Rev. 01A/Dec. 1995)	Contents page at front of every volume indicates revision level of set.
6.3-18 6.3-73	6.3-18 6.3-73	Added "Safety Related" to differentiate between the 250 Vdc Non-Safety Related Battery System per: M12-2(3)-92-005A
Figure 6.3-2A Rev. YE Figure 6.3-2B Rev. BC Figure 6.3-7A Rev. AX Figure 6.3-7B Rev. UM Figure 6.3-9A Rev. AS Figure 6.3-9B Rev. BB	Figure 6.3-2A Rev. YL Figure 6.3-2B Rev. BG Figure 6.3-7A Rev. BB Figure 6.3-7B Rev. UR Figure 6.3-9A Rev. AX Figure 6.3-9B Rev. BH	Insert the latest drawing revisions.
6.4-4 6.4-5	6.4-4 6.4-5	Clarified description of the 2/3 Control Room Ventilation System during normal and emergency pressurized modes of operation.
Table 6.5-1 Sheet 1 of 1	Table 6.5-1 Sheet 1 of 1	Changed units to inches of H_20 rather than feet of H_20 it was previously labeled incorrectly.. Changed per: FSAR Q & A B.10
Figure 6.5-1 Rev. PU	Figure 6.5-1 Rev. PY	Insert the latest drawing revision.
Figure 6.5-2	Figure 6.5-2	SBGT carbon adsorber trays were replaced in design change P12-0-91-694. Figure reflects new tray.
Figure 6.5-3	Figure 6.5-3	Changed units to inches of H_20 for Standby Gas Treatment System Exhaust Fan Static Pressure. It was labeled incorrectly as psid.
Table 7.2-1 Sheet 1 of 1	Table 7.2-1 Sheet 1 of 1	Changed Condenser Low Vacuum Scram Setpoint from 23 in. to 21 in. per Tech Spec Amendments #134 and #128 for Units 2 & 3 respectively.
Table 7.3-1 Sheet 1 of 1	Table 7.3-1 Sheet 1 of 1	Changed HPCI Steam Line High Flow from "150 in H_20 differential to "Less than or Equal to 300% Rated Steam Flow"
7.6-20 7.6-21 7.6-22	7.6-20 7.6-21 7.6-22 7.6-22-a	Added text to Section 7.6.2.2.1 to incorporate minor plant change P12-2-91-698 "Reactor Vessel Shell and Flange Thermocouple Replacement" and Section 7.6.2.2.3 to incorporate Modification M12-2(3)-93-004 "RVWLIS Backfill" changes.

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Contents Page (Revision 0) (For Volume 5)	Contents Page (Rev. 01A/Dec. 1995)	Contents Page at front of every volume indicates revision level of set.
8.1-2	8.1-2	Revised paragraph in Section 8.1.3 to differentiate between the Safety Related and Non-Safety Related 250 Vdc Power System per: M12-2(3)-92-005A
8.3-4	8.3-4	Update Section 8.3.1.2 4160V System to include 27N-R Relays per the second level undervoltage relay replacement for Units 2 & 3.
8.3-5	8.3-5	Updated Section 8.3.1.2.1 System Description to include reference to Buses 23-1 and 33-1 second manual crosstie connection between Unit 2 and Unit 3 per M12-0-91-018 partials A & B.
8.3-8 8.3-9 8.3-19 8.3-20 8.3-21 8.3-24 8.3-25 8.3-26 8.3-27	8.3-8 8.3-9 8.3-19 8.3-20 8.3-21 8.3-21-a 8.3-24 8.3-25 8.3-26 8.3-27	Updated Section 8.3 in multiple areas to include reference to the Non-Safety Related 250 Vdc Battery System per M12-2(3)-92-005A.
Table 8.3-1 Sheet 1 of 7 Table 8.3-1 Sheet 4 of 7 Table 8.3-1 Sheet 7 of 7	Table 8.3-1 Sheet 1 of 7 Table 8.3-1 Sheet 4 of 7 Table 8.3-1 Sheet 7 of 7	Changed 4160 kV Buses 23-1 and 24-1 Bus Tie from 3 to 4 - AMH 4.16 - 250 - MVA, 1200 A per M12-0-91-018 A & B on Sheet 1 of 7. Added "Safety Related" to the 2-250-V Battery Chargers reference on Sheet 4 of 7 per M12-2(3)-92-005A, and added notes to Sheet 7 of 7 reflecting the upgrade of Bus 33 from a 250 MVA to a 350 MVA rating per M12-0-91-019F.
Table 8.3-8 Sheet 1 of 1	Table 8.3-8 Sheet 1 of 1	Added "Safety-Related" to 250 Vdc System Table Reference per M12-2(3)-92-005A
Figure 8.3-1 Rev. G Figure 8.3-2 Rev. AG Figure 8.3-9 Rev. C Figure 8.3-10 Rev. B	Figure 8.3-1 Rev. L Figure 8.3-2 Rev. AH Figure 8.3-9 Rev. D Figure 8.3-10 Rev. C	Insert the latest drawing revisions.
9-i 9-ii 9-iii	9-i 9-ii 9-iii	Revise Table of Contents.
9-vi 9-vii	9-vi 9-vii	Revise list of figures

PAGE CHANGE INDEX / SUMMARY OF CHANGES

PAGE CHANGE INSTRUCTIONS		SUMMARY OF CHANGES
Remove Page (s)	Insert Page (s)	
9.1-3 9.1-5 9.1-11 9.1-15 9.1-16 9.1-17 9.1-21 9.1-22	9.1-3 9.1-5 9.1-5-a 9.1-11 9.1-11-a 9.1-15 9.1-16 9.1-17 9.1-21 9.1-22	Sections 9.1.2.2.2 and 9.1.4.2.3 updated to include Reactor Steam Dryer Removal and Installation changes to procedures DMP 0200-13 and DMP 0200-14 (pages 9.1-3, 9.1-21) Revised Section 9.1.2.2.3.2 Title and added new Section 9.1.2.2.3.3 Spent Fuel Pool Blade Guide Racks and 9.1.2.3.3 Spent Fuel Blade Guide Racks per M12-2(3)-84-120. (pages 9.1-5, 9.1-5a, 9.1-11) Revised pages to include the Refueling Platform Replacement changes per P12-2-93-280 and P12-3-93-273 (pages 9.1-15, 9.1-16, 9.1-17, 9.1-22)
Table 9.1-1 Sheet 1 of 1	Table 9.1-1 Sheet 1 of 1	Updated storage equipment references to include Spent Fuel Pool High Density Racks and Spent Fuel Pool Blade Guide Storage Racks per M12-2(3)-84-120 and added Pole Handling System and Mast Mounted Camera reference to Servicing Aids per P12-2-93-280 and P12-3-93-273. (Delete obsolete Figure 9.1-15)
Figure 9.1-3 Rev. AR Figure 9.1-4 Rev. AD Figure 9.1-13 Rev. AD Figure 9.1-14 Rev. V	Figure 9.1-3 Rev. AT Figure 9.1-4 Rev. AG Figure 9.1-13 Rev. AE Figure 9.1-14 Rev. X	Insert the latest drawing revisions.
Figure 9.1-15	N/A	Figure 9.1-15 Deleted.
9.2-19	9.2-19	Updated Section 9.2.6.3 Safety Evaluation to include Unit 2 & 3 480 Vdc power reference per M12-2(3)-90-057 partials.
Figure 9.2-1 Rev. S Figure 9.2-2 Rev. P Figure 9.2-3 Rev. BN Figure 9.2-4 Rev. NV Figure 9.2-8 Rev. KQ Figure 9.2-9 Rev. AG Figure 9.2-10 Rev. CB Figure 9.2-11 Rev. A Figure 9.2-12 Rev. A Figure 9.2-15 Rev. A Figure 9.2-16 Rev. A Figure 9.2-19 Rev. AJ Figure 9.2-20 Rev. LU Figure 9.2-21 Rev. AD Figure 9.2-22 Rev. AH	Figure 9.2-1 Rev. Y Figure 9.2-2 Rev. W Figure 9.2-3 Rev. BY Figure 9.2-4 Rev. PC Figure 9.2-8 Rev. KS Figure 9.2-9 Rev. AH Figure 9.2-10 Rev. CH Figure 9.2-11 Rev. B Figure 9.2-12 Rev. B Figure 9.2-15 Rev. C Figure 9.2-16 Rev. B Figure 9.2-19 Rev. AN Figure 9.2-20 Rev. LX Figure 9.2-21 Rev. AF Figure 9.2-22 Rev. AL	Insert the latest drawing revisions.

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Remove Page (s)	Insert Page (s)	
9.3-1 9.3-2 9.3-3 9.3-4 9.3-5 9.3-6 9.3-7 9.3-8 9.3-9 9.3-10 9.3-11 9.3-12 9.3-13 9.3-14 9.3-15 9.3-16 9.3-17 9.3-18 9.3-19 9.3-20 9.3-21 9.3-22 9.3-23 9.3-24 9.3-25 9.3-26 9.3-27 9.3-28 9.3-29 9.3-30	9.3-1 9.3-2 9.3-3 9.3-4 9.3-4-a 9.3-5 9.3-6 9.3-7 9.3-8 9.3-9 9.3-10 9.3-11 9.3-12 9.3-13 9.3-14 9.3-15 9.3-16 9.3-17 9.3-18 9.3-19 9.3-20 9.3-21 9.3-22 9.3-23 9.3-24 9.3-25 9.3-26 9.3-27 9.3-28 9.3-29 9.3-30 9.3-31 9.3-32 9.3-33 9.3-34 9.3-35 9.3-36	Multiple changes to Section 9.3 including total re-write of 9.3.2.1 High Radiation Sampling System to reflect the Post TMI NUREG 0737 & R.G 1.97 requirements. Revised text reflects as built differences from preliminary design. Submitted per NRC Commitment Reduction Plan, other changes include incorporation of P12-2-94-265 changes to 9.3.5.2 and 9.3.5.4, and P12-2-93-205 changes to 9.3.1.3.1.
Figure 9.3-3 Rev. AN Figure 9.3-4 Rev. E	Figure 9.3-3 Rev. AQ Figure 9.3-4 Rev. G	Insert the latest drawing revisions.
Figure 9.3-5	Figure 9.3-5	Figure updated per P12-2-93-205 changes and P12-2-94-265 changes.
Figure 9.3-8 Rev. HM	Figure 9.3-8 Rev. HQ	Insert the latest drawing revisions.
9.4-6 9.4-7	9.4-6 9.4-7	Editorial Change

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Remove Page (s)	Insert Page (s)	
Figure 9.4-1 Rev. B Figure 9.4-2 Rev. 0 Figure 9.4-3 Rev. 7 Figure 9.4-4 Rev. AA Figure 9.4-7 Rev. D Figure 9.4-8 Rev. E Figure 9.4-9 Rev. M Figure 9.4-10 Rev. G Figure 9.4-11 Rev. F Figure 9.4-12 Rev. E Figure 9.4-13 Rev. D Figure 9.4-14 Rev. D Figure 9.4-15 Rev. M Figure 9.4-16 Rev. L Figure 9.4-17 Rev. E	Figure 9.4-1 Rev. C Figure 9.4-2 Rev. E Figure 9.4-3 Rev. 10 Figure 9.4-4 Rev. AC Figure 9.4-7 Rev. E Figure 9.4-8 Rev. F Figure 9.4-9 Rev. P Figure 9.4-10A Rev. B Figure 9.4-10B Rev. A Figure 9.4-11 Rev. G Figure 9.4-12 Rev. F Figure 9.4-13 Rev. E Figure 9.4-14 Rev. E Figure 9.4-15 Rev. Q Figure 9.4-16 Rev. Q Figure 9.4-17 Rev. F	Insert the latest drawing revisions.
9.5.1	9.5.1	Added paragraph to Section 9.5.2.2 to incorporate P12-0-93-201 changes on Cellular Phone Antenna Installation
9.5-3 9.5-4	9.5-3 9.5-3-a 9.5-4	Re-write of Section on Intraplant Radio Communication to incorporate Minor Plant Change P12-0-92-603 Completion of 900 MHz Radio Installation.
9.5-8	9.5-8	Revised DGCW Flow Requirements per Ler 2-93-018.
Figure 9.5-1 Rev. M	Figure 9.5-1 Rev. Q	Insert the latest drawing revision.
Contents Page (Revision 0) (For Volume 6)	Contents Page (Rev. 01A/Dec. 1995)	Contents page at front of every volume indicates revision level of set.
10.2-2	10.2-2 10.2-2-a	Deleted old paragraph on FAS Subsystem and replaced with new re-writes for Unit 2 and Unit 3 per P12-3-93-249, D-3 EHC Tubing Upgrade.
Figure 10.3-1 Rev. NP Figure 10.3-2 Rev. AAE Figure 10.3-3 Rev. AG Figure 10.3-4 Rev. NV	Figure 10.3-1 Rev. NS Figure 10.3-2 Rev. AAH Figure 10.3-3 Rev. AJ Figure 10.3-4 Rev. NX	Insert the latest drawing revisions.
10.4-3	10.4-3	Updated Section 10.4.1.5 to incorporate Tech. Spec. Amendments 134 Unit 2 and 138 Unit 3 Condenser Low Vacuum Scram Setpoint change.
Figure 10.4-1 Rev. JY Figure 10.4-7 Rev. X Figure 10.4-8 Rev. KT Figure 10.4-9 Rev. AW Figure 10.4-10 Rev. AA	Figure 10.4-1 Rev. JY Figure 10.4-7 Rev. AB Figure 10.4-8 Rev. KV Figure 10.4-9 Rev. AX Figure 10.4-10 Rev. AB	Insert the latest drawing revisions.

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PAGE CHANGE INSTRUCTIONS		SUMMARY OF CHANGES
Remove Page (s)	Insert Page (s)	
	Verify: 11-i (original T.O.C.)	Verify 11-i, T.O.C. is original. If other than original, i.e., Rev. 01 / Dec. 1995, replace with the copy provided.
11.1-3 11.1-10	11.1-3 11.1-10	Updated text to reflect the existing noncontaminated drains within the RCA and associated administrative controls per LER 237-93-022.
11.2-18 11.2-20	11.2-18 11.2-20	Updated text to reflect the existing noncontaminated drains within the RCA and associated administrative controls per LER 237-93-022.
Figure 11.2-1 Rev. BN Figure 11.2-2 Original Figure 11.2-3 Rev. YF Figure 11.2-4 Rev. MV Figure 11.2-5 Rev. Y Figure 11.2-7 Rev. V Figure 11.2-8 Rev. V Figure 11.2-10 Rev. AJ Figure 11.2-11 Rev. AC Figure 11.2-12 Rev. W Figure 11.2-13 Rev. S Figure 11.2-14 Rev. T Figure 11.2-16 Rev. C Figure 11.2-19 Rev. G Figure 11.2-20 Rev. H	Figure 11.2-1 Rev. BU Figure 11.2-2 Rev. UW Figure 11.2-3 Rev. YJ Figure 11.2-4 Rev. MX Figure 11.2-5 Rev. Z Figure 11.2-7 Rev. W Figure 11.2-8 Rev. X Figure 11.2-10 Rev. AM Figure 11.2-11 Rev. AE Figure 11.2-12 Rev. X Figure 11.2-13 Rev. V Figure 11.2-14 Rev. W Figure 11.2-16 Rev. E Figure 11.2-19 Rev. K Figure 11.2-20 Rev. K	Insert the latest drawing revisions.
Figure 11.2-21	Figure 11.2-21	Revised figure to incorporate Rad Waste Modification M12-2/3-87-002 and changed the Waste Surge Tank to a River Discharge Tank.
Figure 11.3-1 Rev. BL Figure 11.3-4 Rev. HL Figure 11.3-6 Rev. W Figure 11.3-7 Rev. AAE Figure 11.3-8 Rev. NV Figure 11.3-12 Rev. 5 Figure 11.3-13 Rev. AG Figure 11.3-15 Rev. V Figure 11.3-18 Rev. C Figure 11.3-19 Rev. K Figure 11.4-1 Rev. AG	Figure 11.3-1 Rev. BM Figure 11.3-4 Rev. HN Figure 11.3-6 Rev. X Figure 11.3-7 Rev. AAH Figure 11.3-8 Rev. NX Figure 11.3-12 Rev. 6 Figure 11.3-13 Rev. AJ Figure 11.3-15 Rev. X Figure 11.3-18 Rev. D Figure 11.3-19 Rev. M Figure 11.4-1 Rev. AJ	Insert the latest drawing revisions.
11.5-9 11.5-10	11.5-9 11.5-10	Updated Section 11.5.2.3.1 Sping Monitoring Instrumentation to accurately reflect the current methods used for monitoring per DRS 2000-03.
11.5-14	11.5-14 11.5-14-a	Updated Section 11.5.2.7 to reflect the Service Water Radiation Monitoring System Upgrade for Unit 2 P12-2-94-218.

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Remove Page (s)	Insert Page (s)	
Table 11.5-1 Sheet 4 of 4	Table 11.5-1 Sheet 4 of 4	Note added to Table Equipment Alarm Types
12.1-1 12.1-2	12.1-1 12.1-2	Eliminated reference to the (CAC) Corporate ALARA Committee per DAP 12-07.
Contents Page (Revision 0) (For Volume 7)	Contents Page (Rev. 01A/Dec. 1995)	Contents Page at front of every volume indicates revision level of set.
13-i 13-ii 13-iii 13-iv 13-v	13-i 13-ii 13-iii 13-iv	Revise T.O.C., List of tables and figures.
13.1-1 13.1-2 13.1-3 13.1-4 13.1-5 13.1-6 13.1-7 13.1-8 13.1-9 13.1-10 13.1-11 13.1-12 13.1-13	13.1-1 13.1-2	Removed corporate and station specific functions and responsibilities for personnel from section 13.1.
Table 13.1-1 Sheet 1 Figure 13.1-1 Figure 13.1-2	N/A	Removed corporate and station specific functions and responsibilities from section 13.1.
13.2-2	13.2-2	Updated paragraph in Section 13.2.1.1.1 to incorporate procedure changes to DRP 5000-05 and DAP 12-35.
13.3-1 13.3-2 13.3-3 13.3-4 13.3-5 13.3-6 13.3-7 13.3-8 13.3-9 13.3-10	13.3-1	Deletion of details already in the G.S.E.P. removal of redundant information.
13.5-8	13.5-8	Revised paragraph in Section 13.5.2.2.15 to incorporate changes to DAP 9-9 Rev. 8 Special Procedures.

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Remove Page (s)	Insert Page (s)	
15.2-10	15.2-10	Changed Item C in Section 15.2.5.2 to scram at 21 in. Hg Vacuum per upgraded Tech Spec amendments DPR 19 Amend. 134 and DRP 25 Amen. 128.
15.4-14	15.4-14	Change in main condenser vacuum scram/trip setpoint from 23 in Hg to 21 in. Hg per upgraded Tech Spec Amendments DPR 19 Amend. 134 and DPR 25 Amend. 128.

Commonwealth Edison Company
Dresden Generating Station
6500 North Dresden Road
Morris, IL 60450
Tel 815-942-2920

ComEd

February 15, 1996

JSP Ltr: #96 - 0015

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

Subject: Dresden Nuclear Power Station Units 2 and 3
Updated Final Safety Analysis Report, Revision
NRC Docket Numbers 50-237 and 50-249

Reference: 1) B Rybak letter to NRC Document Control Desk, dated December 15, 1995 transmitting UFSAR, Revision 1.

Reference 1 transmitted the biennial update of the Dresden Station UFSAR to the Document Control Desk in accordance with 10 CFR 50.71. This letter is to notify you that we have identified administrative errors in the Dresden Station UFSAR, revision 1 submittal.

We have completed an audit of the UFSAR package and are in process of correcting the administrative errors. A corrected version of the package will be submitted to the Document Control Desk no later than March 31, 1996.

Very truly yours,

J. Stephen Perry
Vice President
BWR Operations

cc: H. J. Miller, Regional Administrator, NRC, Region III
J. F. Stang, Project Manager, NRR (Unit 2/3)
C. L. Vanderniet, Senior Resident Inspector, Dresden
Office Of Nuclear Facility Safety, IDNS

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Commonwealth Edison Company
1400 Opus Place
Downers Grove, IL 60515

ComEd

December 15, 1995

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

Subject: Dresden Nuclear Power Station Units 2 and 3
UFSAR Update Revision 1
NRC Docket No's. 50-237 and 50-249

Reference: P.L. Piet letter to J.B. Martin dated December 30, 1993

The referenced letter documented the completion of the Dresden Station UFSAR rebaseline project. This letter transmits the biennial update (Revision 1) to the Dresden UFSAR, submitted in accordance with 10 CFR 50.71(e). The changes included in Revision 1 are changes to the facility and its procedures and are current through June 30, 1995.

This revision accurately represents changes made since the referenced submittal, as necessary, to reflect information and analyses or prepared pursuant to Commission requirement and also represents changes made under the provisions of 10 CFR 50.59.

Attached to this letter is a detailed change log which identifies and explains all changes from the rebaselined Dresden UFSAR. This is being provided as an aid to the Staff to assist in the review of the changes made.

Pursuant to 10 CFR 50.4 (b) (6), one (1) signed original and ten (10) copies are being provided to the Document Control Desk, plus one (1) copy to the NRC Region III office and one (1) copy to the Dresden Senior Resident Inspector office.

A053
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To the best of my knowledge and belief, the statements contained above are true and correct. In some respect these statements are not based on my personal knowledge, but obtained information furnished by other Commonwealth Edison employees, contractor employees, and consultants. Such information has been reviewed in accordance with Company practice, and I believe it to be reliable.

Please address any questions or comments regarding this submittal to this office.

Sincerely,

OFFICIAL SEAL
MARY JO YACK
NOTARY PUBLIC STATE OF ILLINOIS
MY COMMISSION EXPIRES 11/29/97

Bob Rybak
Nuclear Licensing Administrator

Mary Jo Yack 12-15-95

cc: H.B. Miller, Regional Administrator, RIII
J.F. Stang, Project Manager, NRR
C.L. Vanderniet, Senior Resident Inspector, Dresden
Office of Nuclear Facility Safety, IDNS

REVISIONS CHANGE SUMMARY REV. 01

Identification Of Page(s)	Remove	Insert	Notes
List of Effective Pages Current Throughout Rev. 01	All (Revision 0)	Pages 1 through 23 (Revision 01/Dec. 1995)	List of Effective Pages showing Rev. 01 changes.
T 1.1-1	Table 1.1-1 Sheet 3 of 3 (Original Text)	Table 1.1-1 Sheet 3 of 3 (Rev. 01/Dec. 1995)	Added Reference of Acronym RVWLIS per addition of the RVWLIS Backfill Modification M12-2(3)-93-004
1.2-14	1.2-14 (Original Text)	1.2.14 (Rev. 01/Dec. 1995) Text	Updated description of Refueling Operations to incorporate changes to procedures DMP 0200-13 and DMP 0200-14
1.2-17	1.2-17 (Original Text)	1.2-17 (Rev. 01/Dec. 1995) Text	Updated paragraph to include revised text to incorporate Gaseous Monitoring System and Fuel Storage Building Ventilation Modification M12-0-91-007
1.2-19	1.2-19 (Original Text)	1.2-19 (Rev. 01/Dec 1995) Text	Updated paragraph to include revised text to incorporate Exempt Plant Change Replacement of Valve 2-4608 PCV-4601, P12-2-94-265
F.1.2-2 F.1.2-3 F.1.2-4 F.1.2-9 F.1.2-13	Figure 1.2-2 Rev. F Figure 1.2-3 Rev. E Figure 1.2-4 Rev. K Figure 1.2-9 Rev. A Figure 1.2-13 Rev. A	Figure 1.2-2 Rev. G Figure 1.2-3 Rev. F Figure 1.2-4 Rev. L Figure 1.2-9 Rev. B Figure 1.2-13 Rev. B	Insert Latest Revision

F.1.7-1 F.1.7-2	Figure 1.7-1 Rev. G Figure 1.7-2 Rev. Q	Figure 1.7-1 Rev. H Figure 1.7-2 Rev. R	Insert Latest Revision
2.4-1 2.4-3 T.2.4-1	2.4-1 (Original Text) 2.4-3 (Original Text) Table 2.4-1 Sheet 1 of 1 (Original Text)	2.4-1 (Rev. 01/Dec. 1995) Text 2.4-3 (Rev. 01/Dec. 1995) Text Table 2.4-1 (Rev. 01/Dec. 1995)	Correct text to show maximum historical flood elevation. Remove reference to nominal flood level. Clarify flood values, high river values, and remove incorrect max flood values in Table.
3.1-1	3.1-1 (Original Text)	3.1-1 (Rev. 01/Dec. 1995) Text	Revised paragraph per replacement of valve 2-4608-PCV-4601 in the Unit 2 Air System P12-2-94-265
3.3-8	3.3-8 (Original Text)	3.3-8 (Rev. 01/Dec. 1995) Text	Revised paragraph to incorporate the Isolation Condenser Upgrade Modification M12-2(3)-90-057 changes
3.7-3	3.7-3 (Original Text)	3.7-3 (Rev. 01/Dec. 1995) Text	Updated paragraph to incorporate refueling platform replacement exempt plant changes P12-2-93-280 and P12-3-93-273
3.8-7-a	N/A	3.8-7-a (Rev. 01/Dec. 1995) Text	Addition of text to incorporate RVWLIS Backfill Modification M12-2(3)-93-004 changes

T.3.8-2	Table 3.8-2 Sheet 1 of 1 (Original)	Table 3.8-2 Sheet 1 of 1 (Rev. 01/Dec. 1995)	Revised table to incorporate new penetrations for 3 seperate minor plant changes P12-3-92-714: Split Penetration X-11A into A & B and 111A changed to a type 1A; P12-3-92-715: changed penetration X-138 to type 1A, and P12-3-92-716 Split Penetration X-149 into A & B and changed B to a type 1A
T.3.8-4	Table 3.8-4 Sheet 3 of 3 (Original)	Table 3.8-4 Sheet 3 of 3 (Rev. 01/Dec. 1995)	Updated table to incorporate the RVWLIS Backfill Modification changes to Note #2 to show reference to Unit 2 and Unit 3 and correct RVWLIS acrynym
4.4-6	4.4-6 (Original Text)	4.4-6 (Rev. 01/Dec. 1995) Text	Revised paragraph to show 70% Flow Control Line and deleted last sentence per Commitment to Dresden Response to GL 94-02
F.4.4-1	Figure 4.4-1 (Original)	Figure 4.4-1 (Rev. 01/Dec. 1995)	Revised Figure to show 70% Flow Control Line per Commitment to Dresden Response to GL 94-02
4.6-8 4.6-9	4.6-8 (Original Text) 4.6-9 (Original Text)	4.6-8 (Rev. 01/Dec. 1995) Text 4.6-9 (Rev. 01/Dec. 1995) Text	Added Reference E to Section 4.6.3.3.1 and updated 2nd paragraph in Section 4.6.3.3.2 to incorporate RVWLIS Backfill Modification M12-2(3)-93-004
F.4.6-4 F.4.6-5	Figure 4.6-4 Rev. BB Figure 4.6-5 Rev. AAS	Figure 4.6-4 Rev. BE Figure 4.6-5 Rev. AAU	Insert Latest Revision

5.2-2 5.2-2-a	5.2-2 (Original Text) N/A	5.2-2 (Rev. 01/Dec. 1995) Text 5.2-2-a (Rev. 01/Dec. 1995) Text	Added paragraph to Section 5.2.2.2 that references NUREG 0737, Item II.D.1 Additional Evaluation of Relief & Safety Valve Testing
5.2-14	5.2-14 (Original Text)	5.2-14 (Rev. 01/Dec. 1995) Text	Updated paragraph to incorporate RWCU Pipe Replacement Schedular Commitment to the NRC
5.4-27 5.4-28 5.4-29	5.4-27 (Original Text) 5.4-28 (Original Text) 5.4-29 (Original Text)	5.4-27 (Rev. 01/Dec. 1995) Text 5.4-28 (Rev. 01/Dec. 1995) Text 5.4-29 (Rev. 01/Dec. 1995) Text	Inserted 3 seperate paragraphs to update Sections 5.4.6.2 and 5.4.6.3 which incorporate Isolation Condenser Make-Up Pump Upgrade Modification M12-2(3)-90-057 Partials
5.4-27 5.4-29 5.4-32	5.4-27 (Text) 5.4-29 (Text) 5.4-32 (Original Text)	5.4-27 (Rev. 01/Dec. 1995) Text 5.4-29 (Rev. 01/Dec. 1995) Text 5.4-32 (Rev. 01/Dec. 1995) Text	Added safety related reference to differentiate between the non-safety related 250 VDC Battery System Modification M12-2(3)-92-005a addition
5.4-36	5.4-36 (Original Text)	5.4-36 (Rev. 01/Dec. 1995) Text	Changed RWCU high pressure alarm set from 150 psig to 130 psig per set point change SPC #3-95-022

F.5.4-1 F.5.4-2 F.5.4-3 F.5.4-4 F.5.4-15 F.5.4-18 F.5.4-19 F.5.4-21 F.5.4-23 F.5.4-24 F.5.4-26 F.5.4-27	Figure 5.4-1 Rev. AK Figure 5.4-2 Rev. HQ Figure 5.4-3 Rev. AV Figure 5.4-4 Rev. AJ Figure 5.4-15 Rev. C Figure 5.4-18 Rev. KR Figure 5.4-19 Rev. AM Figure 5.4-21 Rev. AF Figure 5.4-23 Rev. ZK Figure 5.4-24 Rev. AS Figure 5.4-26 Rev. Z Figure 5.4-27 Rev. S	Figure 5.4-1 Rev. AS Figure 5.4-2 Rev. HT Figure 5.4-3 Rev. AZ Figure 5.4-4 Rev. AM Figure 5.4-15 Rev. O Figure 5.4-18 Rev. KW Figure 5.4-19 Rev. AS Figure 5.4-21 Rev. AH Figure 5.4-23 Rev. ZL Figure 5.4-24 Rev. AV Figure 5.4-26 Rev. AA Figure 5.4-27 Rev. T	Insert Latest Revision
6.2-4 6.2-95	6.2-4 (Original Text) 6.2-95 (Original Text)	6.2-4 (Rev. 01/Dec. 1995) Text 6.2-95 (Rev. 01/Dec. 1995) Text	Revised Personnel Airlock Door and Personnel Access Lock Double Door description per: SER/TER info that forwards exemption from certain 10CFR 50.54 (O) & APP. O Requirements
6.2-14 6.2-80 6.2-81	6.2-14 (Original Text) 6.2-80 (Original Text) 6.2-81 (Original Text)	6.2-14 (Rev. 01/Dec. 1995) Text 6.2-80 (Rev. 01/Dec. 1995) Text 6.2-81 (Rev. 01/Dec. 1995) Text	Added RBVS reference to Containment Venting Section and revised the Vent, Purge, and Inerting System Section to show venting the containment during normal operation is allowed
6.2-17	6.2-17 (Original Text)	6.2-17 (Rev. 01/Dec. 1995) Text	Typographical error correction in suppression chamber sizing determination - psia to psig

6.2-70	6.2-70 (Original Text)	6.2-70 (Rev. 01/Dec. 1995) Text	Revised instrument line excess flow check valves sentence to read "and simple" check valves, per RVWLIS Backfill Modification M12-2(3)-93-004
6.2-74 6.2-74-a	6.2-74 (Original Text) N/A	6.2-74 (Rev. 01/Dec. 1995) Text 6.2-74-a (Rev. 01/Dec. 1995) Text	Updated Section 6.2.4.3.2 Containment Integrity to incorporate Minor Plant Changes P12-2-93-220 & P12-3-93-226 upgrade information
6.2-79 6.2-88	6.2-79 (Original Text) 6.2-88 (Original Text)	6.2-79 (Rev. 01/Dec. 1995) Text 6.2-88 (Rev. 01/Dec. 1995) Text	Clarification of ACAD/NCAD description to satisfy a Corrective Action specified in LER 2-95-011

T.6.2-9	Table 6.2-9 Sheets 1 through 10 (Original)	Table 6.2-9 Sheets 1 through 10 (Rev. 01/Dec. 1995)	Multiple changes include: (Sheet 1 of 10) changed penetration X-108A valves 1301-1 and 1301-2 max. iso. times from 30 to 40 seconds. Per: M12-2-92-001 partials C & D. Penetration X-108A valves 1301-17 and 1301-20 changed max. iso. times from 5 to 10 seconds. Per: DATR 3/4.18 and added note to X-106 valves 220-1 and 22-2 showing Unit 3 valves were chaged from gate to globe valves. (Sheet 2 of 10) changed penetration X-109B(A) valves 1301-3 and 1301-4 max. iso. times from 30 to 40 seconds and X-115A(128) valves 2301-4 and 2301-5 max. iso. times from 25 to 50 seconds per M12-2-92-001 partials C, D, E, F & G. Also added note to X-113 valve 1201-1A showing Unit 3 valve was changed from a globe to a gate valve. (Sheet 5 of 10) changed penetration X-147 valve 205-24 max. iso. time from 15 to 45 seconds per: P12-3-93-279
T.6.2-10	Table 6.2-10 Sheets 1 through 3 (Original)	Table 6.2-10 Sheets 1 through 3 (Rev. 01/Dec. 1995)	Changed Drywell Equipment Drains Penetration X-118 valve number from 3-2099-553 to 3-2099-552 per: P12-2-93-220 and P12-3-93-226

T.6.2-11	Table 6.2-11 Sheets 1 through 3 (Original)	Table 6.2-11 Sheets 1 through 3 (Rev. 01/Dec. 1995)	Changed Drywell Equipment Drains Penetration X-118 valve number from 3-2099-553 to 3-2099-552 per P12-2-93-220 and P12-3-93-226
F.6.2-12 F.6.2-13	Figure 6.2-12 Rev. BT Figure 6.2-13 Rev. BB	Figure 6.2-12 Rev. CC Figure 6.2-13 Rev. BE	Insert Latest Revision
6.3-18 6.3-73	6.3-18 (Original Text) 6.3-73 (Original Text)	6.3-18 (Rev. 01/Dec. 1995) Text 6.3-73 (Rev. 01/Dec. 1995) Text	Added "Safety Related" to differentiate between the 250 Vdc Non-Safety Related Battery System per: M12-2(3)-92-005A
F.6.3-2A F.6.3-2B F.6.3-7A F.6.3-7B F.6.3-9A F.6.3-9B	Figure 6.3-2A Rev. YE Figure 6.3-2B Rev. BC Figure 6.3-7A Rev. AX Figure 6.3-7B Rev. UM Figure 6.3-9A Rev. AS Figure 6.3-9B Rev. BB	Figure 6.3-2A Rev. YL Figure 6.3-2B Rev. BG Figure 6.3-7A Rev. BB Figure 6.3-7B Rev. UR Figure 6.3-9A Rev. AX Figure 6.3-9B Rev. BH	Insert Latest Revision
6.4-4 6.4-5	6.4-4 (Original Text) 6.4-5 (Original Text)	6.4-4 (Rev. 01/Dec. 1995) Text 6.4-5 (Rev. 01/Dec. 1995) Text	Clarified description of the 2/3 Control Room Ventilation System during normal and emergency pressurized modes of operation
T.6.5-1	Table 6.5-1 Sheet 1 of 1 (Original)	Table 6.5-1 Sheet 1 of 1 (Rev. 01/Dec. 1995)	Changed units to inches of H_2O rather than feet of H_2O it was previously labeled incorrectly. Changed per: FSAR Q & AB.10
F.6.5-1	Figure 6.5-1 Rev. PU	Figure 6.5-1 Rev. PY	Insert Latest Revision

F.6.5-3	Figure 6.5-3 (Original)	Figure 6.5-3 (Rev. 01/Dec. 1995)	Changed units to inches of H_2O for Standby Gas Treatment System Exhaust Fan Static Pressure. It was labeled incorrectly as psid
T.7.2-1	Table 7.2-1 Sheet 1 of 1 (Original)	Table 7.2-1 Sheet 1 of 1 (Rev. 01/Dec. 1995)	Changed Condenser Low Vacuum Scram Setpoint from 23 in. to 21 in. per Tech Spec Amendments #134 and #128 for Units 2 & 3 respectively
T.7.3-1	Table 7.3-1 Sheet 1 of 1 (Original)	Table 7.3-1 Sheet 1 of 1 (Rev. 01/Dec. 1995)	Changed HPCI Steam Line High Flow from "150 in H_2O differential" to "Less than or Equal to 300% Rated Steam Flow"
7.6-20 7.6-21 7.6-22 7.6-22-a	7.6-20 (Original Text) 7.6-21 (Original Text) 7.6-22 (Original Text) 7.6-22-a N/A	7.6-20 (Rev. 01/Dec. 1995) Text 7.6-21 (Rev. 01/Dec. 1995) Text 7.6-22 (Rev. 01/Dec. 1995) Text 7.6-22-a (Rev. 01/Dec. 1995) Text	Added text to Section 7.6.2.2.1 to incorporate minor plant change P12-2-91-698 "Reactor Vessel Shell and Flange Thermocouple Replacement" and Section 7.6.2.2.3 to incorporate Modification M12-2(3)-93-004 "RVWLIS Backfill" changes
8.1-2	8.1-2 (Original Text)	8.1-2 (Rev. 01/Dec. 1995) Text	Revised paragraph in Section 8.1.3 to differentiate between the Safety Related and Non-Safety Related 250 Vdc Power System per: M12-2(3)-92-005A
8.3-4	8.3-4 (Original Text)	8.3-4 (Rev. 01/Dec. 1995) Text	Update Section 8.3.1.2 4160V System to include 27N-R Relays per the second level undervoltage relay replacement for Units 2 & 3

8.3-5	8.3-5 (Original Text)	8.3-5 (Rev. 01/Dec. 1995) Text	Updated Section 8.3.1.2.1 System Description to include reference to Buses 23-1 and 33-1 second manual crosstie connection between Unit 2 and Unit 3 per M12-0-91-018 partials A & B
8.3-8 8.3-9 8.3-19 8.3-20 8.3-21 8.3-21-a 8.3-24 8.3-25 8.3-26 8.3-27	8.3-8 (Original Text) 8.3-9 (Original Text) 8.3-19 (Original Text) 8.3-20 (Original Text) 8.3-21 (Original Text) 8.3-21-a N/A 8.3-24 (Original Text) 8.3-25 (Original Text) 8.3-26 (Original Text) 8.3-27 (Original Text)	8.3-8 (Rev. 01/Dec. 1995) Text 8.3-9 (Rev. 01/Dec. 1995) Text 8.3-19 (Rev. 01/Dec. 1995) Text 8.3-20 (Rev. 01/Dec. 1995) Text 8.3-21 (Rev. 01/Dec. 1995) Text 8.3-21-a (Rev. 01/Dec. 1995) Text 8.3-24 (Rev. 01/Dec. 1995) Text 8.3-25 (Rev. 01/Dec. 1995) Text 8.3-26 (Rev. 01/Dec. 1995) Text 8,3-27 (Rev. 01/Dec. 1995) Text	Updated Section 8.3 in multiple areas to include reference to the Non-Safety Related 250 Vdc Battery System per M12-2(3)-92-005A
T.8.3-1	Table 8.3-1 Sheet 1 of 7 (Original Text) Table 8.3-1 Sheet 4 of 7 (Original Text) Table 8.3-1 Sheet 7 of 7 (Original Text)	Table 8.3-1 Sheet 1 of 7 (Rev. 01/Dec. 1995) Text Table 8.3-1 Sheet 4 of 7 (Rev. 01/Dec. 1995) Text Table 8.3-1 Sheet 7 of 7 (Rev. 01/Dec. 1995) Text	Changed 4160 kV Buses 23-1 and 24-1 Bus Tie from 3 to 4 - AMH 4.16 - 250 - MVA, 1200 A per M12-0-91-018 A & B on Sheet 1 of 7. Added "Safety Related" to the 2-250-V Battery Chargers reference on Sheet 4 of 7 per M12-2(3)-92-005A, and added notes to Sheet 7 of 7 reflecting the upgrade of Bus 33 from a 250 MVA to a 350 MVA rating per M12-0-91-019F
T.8.3.8	Table 8.3-8 Sheet 1 of 1 (Original Text)	Table 8.3-8 Sheet 1 of 1 (Rev. 01/Dec. 1995) Text	Added "Safety-Related" to 250 Vdc System Table Reference per M12-2(3)-92-005A

F.8.3-1 F.8.3-2 F.8.3-9 F.8.3-10	Figure 8.3-1 Rev. G Figure 8.3-2 Rev. AG Figure 8.3-9 Rev. C Figure 8.3-10 Rev. B	Figure 8.3-1 Rev. L Figure 8.3-2 Rev. AH Figure 8.3-9 Rev. D Figure 8.3-10 Rev. C	Insert Latest Revision
Table of Contents	9-ii (Original T.O.C.) 9-iii (Original T.O.C.)	9-ii (Rev. 01/Dec. 1995) T.O.C. 9-iii (Rev. 01/Dec. 1995) T.O.C.	9.3.2 through 9.3-6 updated
List of Figures	9-vi (Original L.O.F.)	9-vi (Rev. 01/Dec. 1995) L.O.F.	Figure 9.1-15 was deleted
9.1-3 9.1-21	9.1-3 (Original Text) 9.1-21 (Original Text)	9.1-3 (Rev. 01/Dec. 1995) Text 9.1-21 (Rev. 01/Dec. 1995) Text	Sections 9.1.2.2.2 and 9.1.4.2.3 updated to include Reactor Steam Dryer Removal and Installation changes to procedures DMP 0200-13 and DMP 0200-14
9.1-5 9.1-5-a 9.1-11	9.1-5 (Original Text) 9.1-5-a N/A 9.1-11 (Original Text)	9.1-5 (Rev. 01/Dec. 1995) Text 9.1-5-a (Rev. 01/Dec. 1995) Text 9.1-11 (Rev. 01/Dec. 1995) Text	Revised Section 9.1.2.2.3.2 title and added new Section 9.1.2.2.3.3 Spent Fuel Pool Blade Guide Racks and 9.1.2.3.3 Spent Fuel Pool Blade Guide Racks per M12-2(3)-84-120
9.1-15 9.1-16 9.1-17 9.1-22	9.1-15 (Original Text) 9.1-16 (Original Text) 9.1-17 (Original Text) 9.1-22 (Original Text)	9.1-15 (Rev. 01/Dec. 1995) Text 9.1-16 (Rev. 01/Dec. 1995) Text 9.1-17 (Rev. 01/Dec. 1995) Text 9.1-22 (Rev. 01/Dec. 1995) Text	Revised pages to include the Refueling Platform Replacement changes per P12-2-93-280 and P12-3-93-273

T.9.1-1	Table 9.1-1 Sheet 1 of 1 (Original)	Table 9.1-1 Sheet 1 of 1 (Rev, 01/Dec. 1995)	Updated storage equipment references to include Spent Fuel Pool High Density Racks and Spent Fuel Pool Blade Guide Storage Racks per M12-2(3)-84-120 and added Pole Handling System and Mast Mounted Camera reference to Servicing Aids per P12-2-93-280 and P12-3-93-273
F.9.1-3 F.9.1-4 F.9.1-13 F.9.1-14	Figure 9.1-3 Rev. AR Figure 9.1-4 Rev. AD Figure 9.1-13 Rev. AD Figure 9.1-14 Rev. V	Figure 9.1-3 Rev. AT Figure 9.1-4 Rev. AG Figure 9.1-13 Rev. AE Figure 9.1-14 Rev. X	Insert Latest Revision
9.2-19	9.2-19 (Original Text)	9.2-19 (Rev. 01/Dec. 1995) Text	Updated Section 9.2-63 Safety Evaluation to include Unit 2 & 3 480 Vdc power reference per M12-2(3)-90-057 partials

F.9.2-1 F.9.2-2 F.9.2-3 F.9.2-4 F.9.2-8 F.9.2-9 F.9.2-10 F.9.2-11 F.9.2-12 F.9.2-15 F.9.2-16 F.9.2-19 F.9.2-20 F.9.2-21 F.9.2-22	Figure 9.2-1 Rev. S Figure 9.2-2 Rev. P Figure 9.2-3 Rev. BN Figure 9.2-4 Rev. NV Figure 9.2-8 Rev. KQ Figure 9.2-9 Rev. AG Figure 9.2-10 Rev. CB Figure 9.2-11 Rev. A Figure 9.2-12 Rev. A Figure 9.2-15 Rev. A Figure 9.2-16 Rev. A Figure 9.2-19 Rev. AJ Figure 9.2-20 Rev. LU Figure 9.2-21 Rev. AD Figure 9.2-22 Rev. AH	Figure 9.2-1 Rev. X Figure 9.2-2 Rev. W Figure 9.2-3 Rev. BX Figure 9.2-4 Rev. PC Figure 9.2-8 Rev. KS Figure 9.2-9 Rev. AH Figure 9.2-10 Rev. CH Figure 9.2-11 Rev. B Figure 9.2-12 Rev. B Figure 9.2-15 Rev. C Figure 9.2-16 Rev. B Figure 9.2-19 Rev. AN Figure 9.2-20 Rev. LX Figure 9.2-21 Rev. AF Figure 9.2-22 Rev. AL	Insert Latest Revision

9.3-1 9.3-2 9.3-3 9.3-4 9.3-5 9.3-6 9.3-7 9.3-8 9.3-9 9.3-10 9.3-11 9.3-12 9.3-13 9.3-14 9.3-15 9.3-16 9.3-17 9.3-18 9.3-19 9.3-20 9.3-21 9.3-22 9.3-23 9.3-24 9.3-25 9.3-26 9.3-27 9.3-28 9.3-29	9.3-1 (Original Text) 9.3-2 (Original Text) 9.3-3 (Original Text) 9.3-4 (Original Text) 9.3-4-a N/A 9.3-5 (Original Text) 9.3-6 (Original Text) 9.3-7 (Original Text) 9.3-8 (Original Text) 9.3-9 (Original Text) 9.3-10 (Original Text) 9.3-11 (Original Text) 9.3-12 (Original Text) 9.3-13 (Original Text) 9.3-14 (Original Text) 9.3-15 (Original Text) 9.3-16 (Original Text) 9.3-17 (Original Text) 9.3-18 (Original Text) 9.3-19 (Original Text) 9.3-20 (Original Text) 9.3-21 (Original Text) 9.3-22 (Original Text) 9.3-23 (Original Text) 9.3-24 (Original Text) 9.3-25 (Original Text) 9.3-26 (Original Text) 9.3-27 (Original Text) 9.3-28 (Original Text) 9.3-29 (Original Text)	9.3-1 (Rev. 01/Dec. 1995) Text 9.3-2 (Rev. 01/Dec. 1995) Text 9.3-3 (Rev. 01/Dec. 1995) Text 9.3-4 (Rev. 01/Dec. 1995) Text 9.3-4-a (Rev. 01/Dec. 1995) Text 9.3-5 (Rev. 01/Dec. 1995) Text 9.3-6 (Rev. 01/Dec. 1995) Text 9.3-7 (Rev. 01/Dec. 1995) Text 9.3-8 (Rev. 01/Dec. 1995) Text 9.3-9 (Rev. 01/Dec. 1995) Text 9.3-10 (Rev. 01/Dec. 1995) Text 9.3-11 (Rev. 01/Dec. 1995) Text 9.3-12 (Rev. 01/Dec. 1995) Text 9.3-13 (Rev. 01/Dec. 1995) Text 9.3-14 (Rev. 01/Dec. 1995) Text 9.3-15 (Rev. 01/Dec. 1995) Text 9.3-16 (Rev. 01/Dec. 1995) Text 9.3-17 (Rev. 01/Dec. 1995) Text 9.3-18 (Rev. 01/Dec. 1995) Text 9.3-19 (Rev. 01/Dec. 1995) Text 9.3-20 (Rev. 01/Dec. 1995) Text 9.3-21 (Rev. 01/Dec. 1995) Text 9.3-22 (Rev. 01/Dec. 1995) Text 9.3-23 (Rev. 01/Dec. 1995) Text 9.3-24 (Rev. 01/Dec. 1995) Text 9.3-25 (Rev. 01/Dec. 1995) Text 9.3-26 (Rev. 01/Dec. 1995) Text 9.3-27 (Rev. 01/Dec. 1995) Text 9.3-28 (Rev. 01/Dec. 1995) Text 9.3-29 (Rev. 01/Dec. 1995) Text	Multiple changes to Section 9.3 including total re-write of 9.3.2.1 High Radiation Sampling System to reflect the Post TMI NUREG 0737 & R.G 1.97 requirements. Revised text reflects as built differences from preliminary design. Submitted per NRC Commitment Reduction Plan, other changes include incorporation of P12-2-94-265 changes to Section 9.3.[illegible] M12-2(3)-84-119 changes to 9.3.5.2 and 9.3.5.4, and P12-2-93-205 changes to 9.3.1.3.1

9.3-30 9.3-31 9.3-32 9.3-33 9.3-34 9.3-35 9.3-36	9.3-30 (Original Text) 9.3-31 (Original Text) 9.3-32 (Original Text) 9.3-33 (Original Text) 9.3-34 (Original Text) 9.3-35 (Original Text) 9.3-36 (Original Text)	9.3-30 (Rev. 01/Dec. 1995) Text 9.3-31 (Rev. 01/Dec. 1995) Text 9.3-32 (Rev. 01/Dec. 1995) Text 9.3-33 (Rev. 01/Dec. 1995) Text 9.3-34 (Rev. 01/Dec. 1995) Text 9.3-35 (Rev. 01/Dec. 1995) Text 9.3-36 (Rev. 01/Dec. 1995) Text	(Continued)
F.9.3-5	Figure 9.3-5 (Original)	Figure 9.3-5 (Rev. 01/Dec. 1995)	Figure updated per P12-2-93-205 changes and P12-2-94-265 changes
F.9.3-3 F.9.3-4 F.9.3-8	Figure 9.3-3 Rev. AN Figure 9.3-4 Rev. E Figure 9.3-8 Rev. HM	Figure 9.3-3 Rev. AQ Figure 9.3-4 Rev. G Figure 9.3-8 Rev. HQ	Insert Latest Revision
F.9.4-1 F.9.4-2 F.9.4-3 F.9.4-4 F.9.4-7 F.9.4-8 F.9.4-9 F.9.4-10A F.9.4-10B F.9.4-11 F.9.4-12 F.9.4-13 F.9.4-14 F.9.4-15 F.9.4-16 F.9.4-17	Figure 9.4-1 Rev. B Figure 9.4-2 Rev. O Figure 9.4-3 Rev. 7 Figure 9.4-4 Rev. AA Figure 9.4-7 Rev. D Figure 9.4-8 Rev. E Figure 9.4-9 Rev. M Figure 9.4-10 Rev. G Figure 9.4-11 Rev. F Figure 9.4-12 Rev. E Figure 9.4-13 Rev. D Figure 9.4-14 Rev. D Figure 9.4-15 Rev. M Figure 9.4-16 Rev. L Figure 9.4-17 Rev. E	Figure 9.4-1 Rev. C Figure 9.4-2 Rev. E Figure 9.4-3 Rev. 10 Figure 9.4-4 Rev. AC Figure 9.4-7 Rev. E Figure 9.4-8 Rev. F Figure 9.4-9 Rev. P Figure 9.4-10A Rev. Figure 9.4-10B Rev. Figure 9.4-11 Rev. G Figure 9.4-12 Rev. F Figure 9.4-13 Rev. E Figure 9.4-14 Rev. E Figure 9.4-15 Rev. Q Figure 9.4-16 Rev. Q Figure 9.4-17 Rev. F	Insert Latest Revision

9.5-1	9.5-1 (Original Text)	9.5-1 (Rev. 01/Dec. 1995) Text	Added paragraph to Section 9.5.2.2 to incorporate P12-0-93-201 changes on Cellular Phone Antenna Installation
9.5-3 9.5-3-a 9.5-4	9.5-3 (Original Text) 9.5-3-a N/A 9.5-4 (Original Text)	9.5-3 (Rev. 01/Dec. 1995) Text 9.5-3-a (Rev. 01/Dec. 1995) Text 9.5-4 (Rev. 01/Dec. 1995) Text	Re-write of Section on Intraplant Radio Communication to incorporate Minor Plant Change P12-0-92-603 Completion of 900 MHz Radio Installation
9.5-8	9.5-8 (Original Text)	9.5-8 (Rev. 01/Dec. 1995) Text	Revised DGCW Flow Requirements per LER 2-93-018
F.9.5-1	Figure 9.5-1 Rev. M	Figure 9.5-1 Rev. Q	Insert Latest Revision
10.2-2 10.2-2-a	10.2-2 (Original Text) 10.2-2-a N/A	10.2-2 (Rev. 01/Dec. 1995) Text 10.2-2-a (Rev. 01/Dec. 1995) Text	Deleted old paragraph on FAS Subsystem and replaced with new re-writes for Unit 2 and Unit 3 per P12-3-93-249, D-3 EHC Tubing Upgrade
F.10.3-1 F.10.3-2 F.10.3-3 F.10.3-4	Figure 10.3-1 Rev. NP Figure 10.3-2 Rev. AAE Figure 10.3-3 Rev. AG Figure 10.3-4 Rev. NV	Figure 10.3-1 Rev. NS Figure 10.3-2 Rev. AAH Figure 10.3-3 Rev. AJ Figure 10.3-4 Rev. NX	Insert Latest Revision
10.4-3	10.4-3 (Original Text)	10.4-3 (Rev. 01/Dec. 1995) Text	Updated Section 10.4.1.5 to incorporate Tech. Spec. Ammendments 134 Unit 2 and 128 Unit 3 Condenser Low Vacuum Scram Setpoint change

F.10.4-1 F.10.4-7 F.10.4-8 F.10.4-9 F.10.4-10	Figure 10.4-1 Rev. JY Figure 10.4-7 Rev. X Figure 10.4-8 Rev. KT Figure 10.4-9 Rev. AW Figure 10.4-10 Rev. AA	Figure 10.4-1 Rev. JT Figure 10.4-7 Rev. AB Figure 10.4-8 Rev. KV Figure 10.4-9 Rev. AX Figure 10.4-10 Rev. AB	Insert Latest Revision
Table of Contents	11-i (Original T.O.C.)	11-i (Rev. 01/Dec. 1995) T.O.C	Added Page 11.1-3-a to T.O.C.
11.1-3 11.1-10	11.1-3 (Original Text) 11.1-10 (Original Text)	11.1-3 (Rev. 01/Dec. 1995) Text 11.1-10 (Rev. 01/Dec. 1995) Text	Updated text to reflect the existing noncontaminated drains within the RCA and associated administrative controls per LER 237-93-022
11.2-18 11.2-20	11.2-18 (Original Text) 11.2-20 (Original Text)	11.2-18 (Rev. 01/Dec. 1995) Text 11.2-20 (Rev. 01/Dec. 1995) Text	Updated text to reflect the existing noncontaminated drains within the RCA and associated administrative controls per LER 237-93-022
F.11.2-21	Figure 11.2-21 (Original)	Figure 11.2-21(Rev. 01/Dec. 1995)	Revised figure to incorporate Rad Waste Modification M12-2/3-87-002 and changed the Waste Surge Tank to a River Discharge Tank

F.11.2-1 F.11.2-2 F.11.2-3 F.11.2-4 F.11.2-5 F.11.2-7 F.11.2-8 F.11.2-10 F.11.2-11 F.11.2-12 F.11.2-13 F.11.2-14 F.11.2-16 F.11.2-19 F.11.2-20	Figure 11.2-1 Rev. BN Figure 11.2-2 (Original) Figure 11.2-3 Rev. YF Figure 11.2-4 Rev. MV Figure 11.2-5 Rev. Y Figure 11.2-7 Rev. V Figure 11.2-8 Rev. V Figure 11.2-10 Rev. AJ Figure 11.2-11 Rev. AC Figure 11.2-12 Rev. W Figure 11.2-13 Rev. S Figure 11.2-14 Rev. T Figure 11.2-16 Rev. C Figure 11.2-19 Rev. G Figure 11.2-20 Rev. H	Figure 11.2-1 Rev. BU Figure 11.2-2 Rev. UW Figure 11.2-3 Rev. YJ Figure 11.2-4 Rev. MX Figure 11.2-5 Rev. Z Figure 11.2-7 Rev. W Figure 11.2-8 Rev. X Figure 11.2-10 Rev. AM Figure 11.2-11 Rev. AE Figure 11.2-12 Rev. X Figure 11.2-13 Rev. V Figure 11.2-14 Rev. W Figure 11.2-16 Rev. E Figure 11.2-19 Rev. K Figure 11.2-20 Rev. J	Insert Latest Revision
F.11.3-1 F.11.3-4 F.11.3-6 F.11.3-7 F.11.3-8 F.11.3-12 F.11.3-13 F.11.3-15 F.11.3-18 F.11.3-19	Figure 11.3-1 Rev. BL Figure 11.3-4 Rev. HL Figure 11.3-6 Rev. W Figure 11.3-7 Rev. AAE Figure 11.3-8 Rev. NU Figure 11.3-12 Rev. 5 Figure 11.3-13 Rev. AG Figure 11.3-15 Rev. V Figure 11.3-18 Rev. C Figure 11.3-19 Rev. K	Figure 11.3-1 Rev. BM Figure 11.3-4 Rev. HN Figure 11.3-6 Rev. X Figure 11.3-7 Rev. AAH Figure 11.3-8 Rev. NX Figure 11.3-12 Rev. 6 Figure 11.3-13 Rev. AJ Figure 11.3-15 Rev. X Figure 11.3-18 Rev. D Figure 11.3-19 Rev. M	Insert Latest Revision
F.11.4-1	Figure 11.4-1 Rev. AG	Figure 11.4-1 Rev. AJ	Insert Latest Revision
11.5-9 11.5-10	11.5-9 (Original Text) 11.5-10 (Original Text)	11.5-9 (Rev. 01/Dec. 1995) Text 11.5-10 (Rev. 01/Dec. 1995) Text	Updated Section 11.5.2.3.1 Sping Monitoring Instrumentation to accurately reflect the current methods used for monitoring per DRS 2000-03

11.5-14 11.5-14-a	11.5-14 (Original Text) 11.5-14-a N/A	11.5-14 (Rev. 01/Dec. 1995) Text 11.5-14-a (Rev. 01/Dec. 1995) Text	Updated Section 11.5.2.7 to reflect the Service Water Radiation Monitoring System Upgrade for Unit 2 P12-2-94-218
T.11.5-1	Table 11.5-1 Sheet 4 of 4 (Original)	Table 11.5-1 Sheet 4 of 4 (Rev. 01/Dec. 1995)	Note added to Table Equipment Alarm Types
12.1-1 12.1-2	12.1-1 (Original Text) 12.1-2 (Original Text)	12.1-1 (Rev. 01/Dec. 1995) Text 12.1-2 (Rev. 01/Dec. 1995) Text	Eliminated reference to the (CAC) Corporate ALARA Committee per DAP 12-07
Table of Contents	13-i (Original T.O.C.) 13-II (Original T.O.C.)	13-i (Rev. 01/Dec. 1995) T.O.C. 13-ii (Rev. 01/Dec. 1995) T.O.C.	Deletion of details in 13.3-1 through 13.3-9
13.2-2	13.2-2 (Original Text)	13.2-2 (Rev. 01/Dec. 1995) Text	Updated paragraph in Section 13.2.1.1.1 to incorporate procedure changes to DRP 5000-05 and DAP 12-35
13.3-1	13.3-1 (Original Text) 13.3-2 (Original Text) 13.3-3 (Original Text) 13.3-4 (Original Text) 13.3-5 (Original Text) 13.3-6 (Original Text) 13.3-7 (Original Text) 13.3-8 (Original Text) 13.3-9 (Original Text) 13.3-10 (Original References)	13.3-1 (Rev. 01/Dec. 1995) Text	Deletion of details already in the G.S.E.P. removal of redundant information

13.5-8	13.5-8 (Original Text)	13.5-8 (Rev. 01/Dec. 1995) Text	Revised paragraph in Section 13.5.2.2.15 to incorporate changes to DAP 9-9 Rev. 8 Special Procedures
15.2-10	15.2-10 (Original Text)	15.2-10 (Rev. 01/Dec. 1995) Text	Changed Item C in Section 15.2.5.2 to scram at 21 in. Hg Vacuum per upgraded Tech Spec ammendments DPR 19 Amend. 134 and DPR 25 Amen. 128
15.4-14	15.4-14 (Original Text)	15.4-14 (Rev. 01/Dec. 1995) Text	Change in main condenser vacuum scram/trip setpoint from 23 in Hg to 21 in. Hg per upgraded Tech Spec Amendments DPR 19 Amend. 134 and DPR 25 Amend. 128

Commonwealth Edison Company
1400 Opus Place
Downers Grove, IL 60515

August 3, 1995

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

ComEd

Subject: Dresden Nuclear Power Station Units 2 and 3
Quad Cities Nuclear Power Station Units 1 and 2
Withdrawal of Supplemental Technical Specification Application; and
Partial Implementation of Technical Specification Amendments 137 and 131 to Facility Operating Licenses DPR-19 and DPR-25 and Amendments 158 and 154 to Facility Operating Licenses DPR-29 and DPR-30
NRC Docket Nos. 50-237/249 and 50-254/265

Reference: (a) P. Piet (ComEd) letter to U.S. NRC, dated July 29, 1992.

(b) J. Schrage (ComEd) letter to U.S. NRC, dated July 19, 1995.

(c) J. Stang letter to D. Farrar (ComEd), dated July 27, 1995.

The purpose of this letter is to formally withdraw the proposed supplemental amendment request, as presented in the Reference (b) letter for Dresden and Quad Cities Stations. With the approval of Reference (a) by the NRC staff (Reference (c)), the Reference (b) request is no longer necessary. As such, ComEd will implement the new Surveillance Requirement (SR) (Technical Specification Upgrade Program [TSUP] SR 4.3.D.3), as 4.3.C.2, prior to August 20, 1995 for Quad Cities Station and prior to August 11, 1995 for Dresden Station.

In order to appropriately control the implementation of the revised Surveillance Requirements, the revised pages (in the current format) provided in Attachment A to the Reference (b) letter will serve as the controlling documentation. These pages are also provided herein. ComEd will implement the remainder of Amendments 137 and 131 for Dresden Station and Amendments 158 and 154 for Quad Cities Station during the full implementation of the Technical Specification Upgrade Program.

If there are any questions regarding this matter, please contact this office.

Sincerely,

Peter L. Piet
Nuclear Licensing Administrator

Attachment: Revised Technical Specification Pages

cc: H. J. Miller, Regional Administrator - RIII
J. F. Stang, Project Manager - NRR
R. M. Pulsifer, Project Manager - NRR
M. N. Leach, Senior Resident Inspector - Dresden
C. G. Miller, Senior Resident Inspector - Quad Cities
Office of Nuclear Facility Safety - IDNS

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ATTACHMENT

Revised Technical Specification Pages

Dresden Station		**Quad Cities Station**	
DPR-19	**DPR-25**	**DPR-29**	**DPR-30**
3/4.3-11	**3/4.3-11**	**3.3/4.3-6**	**3.3/4.3-4**
		3.3/4.3-7	**3.3/4.3-5**

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3.3 LIMITING CONDITION FOR OPERATION (Cont'd)

2. The maximum scram insertion time for 90 insertion of any operable control rod shall not exceed 7.00 seconds.

D. Control Rod Accumulators

At all reactor operating pressures, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around this rod has a:

1. Inoperable accumulator,

2. Directional control valve electrically disarmed while in a non-fully Inserted position.

4.3 SURVEILLANCE REQUIREMENT (Cont'd)

2. The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than 800 psig and, during single control rod scram time tests, with the control rod drive pumps isolated from the accumulators, for at least 10% of the control rods, on a rotating basis, at least once per 120 days of power operation.

3. Following completion of each set of scram testing as described above, the results shall be compared against the average scram speed distribution used in the transient analysis to verify applicability of the current MCPR Operating Limit. Refer to Specification 3.5.L

D. Control Rod Accumulators

Once a shift check the status of the pressure and level alarms for each accumulator.

3.3 LIMITING CONDITION FOR OPERATION (Cont'd)

2. The maximum scram insertion time for 90 insertion of any operable control rod shall not exceed 7.00 seconds.

D. Control Rod Accumulators

At all reactor operating pressures, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around this rod has a:

1. Inoperable accumulator,

2. Directional control valve electrically disarmed while in a non-fully Inserted position.

4.3 SURVEILLANCE REQUIREMENT (Cont'd)

2. The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than 800 psig and, during single control rod scram time tests, with the control rod drive pumps isolated from the accumulators, for at least 10% of the control rods, on a rotating basis, at least once per 120 days of power operation.

3. Following completion of each set of scram testing as described above, the results shall be compared against the average scram speed distribution used in the transient analysis to verify applicability of the current MCPR Operating Limit. Refer to Specification 3.5.L

D. Control Rod Accumulators

Once a shift check the status of the pressure and level alarms for each accumulator.

a. Close within 30 seconds after receipt of a signal for control rods to scram, and

b. Open when the scram signal is reset.

C. Scram Insertion Times

1. The average scram insertion time, based on the deenergization of the scram pilot valve solenoids at time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Average Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	3.50

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

% Inserted From Fully Withdrawn	Average Scram Times (sec)
5	0.398
20	0.954
50	2.12
90	3.80

2. The maximum scram insertion time for 90% of any operable control rod shall not exceed 7 seconds.

C. Scram Insertion Times

1. After refueling outage and prior to operation above 30% power, with reactor pressure above 800 psig, all control rods shall be subject to scram-time measurements from the fully withdrawn position. The scram times for single rod scram testing shall be measured without reliance on the control rod drive pumps.

2. The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than 800 psig and, during single control rod scram time tests, with the control rod drive pumps isolated from the accumulators, for at least 10% of the control rods, on a rotating basis, at least once per 120 days of power operation.

3. If Specification 3.3.C.1 cannot be met, the reactor shall not be made supercritical; if operating, the reactor shall be shut down immediately upon determination that average scram time is deficient.

4. If Specification 3.3.C.2 cannot be met, the deficient control rod shall be considered inoperable, fully inserted into the core, and electrically disarmed.

5. If the overall average of the 20% insertion scram time data generated to date in the current cycle exceeds the limit specified in the CORE OPERATING LIMITS REPORT, the MCPR operating limit must be modified as required by Specification 3.5.K.

3. The cycle cumulative mean scram time for 20% insertion will be determined immediately following the testing required in Specifications 4.3.C.1 and 4.3.C.2 and the MCPR operating limit adjusted, if necessary, as required by Specification 3.5.K.

D. Control Rod Accumulators

At all reactor operating pressures, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around that rod has:

1. an inoperable accumulator,

D. Control Rod Accumulators

Once a shift, check the status of the pressure and level alarms for each accumulator.

c. the operating power level shall be limited so that the MCPR will remain above the MCPR fuel cladding integrity safety limit assuming a single error that results in complete withdrawal of any single operable control rod.

C. Scram Insertion Times

1. The average scram insertion time. Based on the deenergization of the scram pilot valve solenoids at time zero of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Average Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	3.50

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

% Inserted From Fully Withdrawn	Average Scram Times (sec)
5	0.398
20	0.954
50	2.12
90	3.80

2. The maximum scram insertion time for 90% insertion of any operable control rods shall not exceed 7 seconds.

3. If Specification 3.3.C.1 cannot be met the reactor shall not be made supercritical: if operating the reactor shall be shut down immediately upon determination that average scram time is deficient.

4. If Specification 3.3.C.2 cannot be met. The deficient control rod shall be con-

C. Scram Insertion Times

1. After refueling outage and prior to operation above 30% power, with reactor pressure above 800 psig, all control rods shall be subject to scram-time measurements from the fully withdrawn position. The scram times for single rod scram testing shall be measured without reliance on the control rod drive pumps.

2. The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than 800 psig and, during single control rod scram time tests, with the control rod drive pumps isolated from the accumulators, for at least 10% of the control rods, on a rotating basis, at least once per 120 days of power operation.

sidered inoperable, fully inserted into the core and electrically disarmed.

5. If the overall average of the 20% insertion scram time data generated to date in the current cycle exceeds the limit specified in the CORE OPERATING LIMITS REPORT, the MCPR operating limit must be modified as required by Specification 3.5.K.

5. The cycle cumulative mean scram time for 20% insertion will be determined immediately following the testing required in Specifications 4.3.C.1 and 4.3.C.2 and the MCPR operating limit adjusted, if necessary, as required by Specification 3.5.K.

D. Control Rod Accumulators

At all reactor operating pressures, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around that rod has:

1. An inoperable accumulator,

2. A directional control valve electrically disarmed while in a nonfully inserted position, or

3. A scram insertion greater than maximum permissible insertion time.

If a control rod with an inoperable accumulator is inserted full-in and its directional control valves are electrically disarmed, it shall not be considered to have an inoperable accumulator, and the rod block associated with that inoperable accumulator may be bypassed.

D. Control Rod Accumulators

Once a shift, check the status of the pressure and level alarms for each accumulator.

E. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δ k. If this limit is exceeded, the reactor shall be shutdown until the cause has been determined and corrective actions have been taken. In accordance with Specification 6.6, the NRC shall be notified of this reportable occurrence within 24 hours.

E. Reactivity Anomalies

During the startup test program and startups following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every equivalent full power month.

F. Economic Generation Control System

Operation of the unit with the economic generation control system with automatic flow control shall be permissible only in the range of 65% to 100% of rated core flow, with reactor power above 20%.

F. Economic Generation Control System

Prior to entering EGC and once per shift while operating in EGC, the EGC operating parameters will be reviewed for acceptability.

PACKAGE DIVIDER

DRP-003