Commonwealth Edison Company
1400 Opus Place
Downers Grove, IL 60515-5701

August 29, 1997

ComEd

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: Quad Cities Nuclear Power Station Units 1 and 2, Dresden Nuclear Power Station Units 2 and 3, LaSalle County Nuclear Power Station Units 1 and 2, Application for Amendment Request to Facility Operating Licenses , DPR-29 and DPR-30, DPR-19 and DPR-25, and NPF-11 and NPF-18, respectively, **Technical Specification Changes for Transition to Siemens Power Corporation ATRIUM-9B Fuel**, Docket Nos. 50-254 and 50-265, 50-237 and 50-249, and 50-373 and 50-374, respectively.

References: See Attachment I

Purpose

Pursuant to 10 CFR 50.90, ComEd proposes to amend Appendix A, Technical Specifications of Facility Operating Licenses DPR-29, DPR-30, DPR-19, DPR-25, NPF-11, and NPF-18 to reflect additional changes due to the transition to Siemens Power Corporation (SPC) ATRIUM-9B fuel. In summary, the proposed changes incorporate: a) new Siemens' methodologies that will enhance operational flexibility and reduce the likelihood of future plant derates, b) administrative changes that both eliminate the cycle specific implementation of ATRIUM-9B fuel and adopt Improved Technical Specification language, where appropriate, and c) changes to the Dresden and Quad Cities Minimum Critical Power Ratio (MCPR) Safety Limits.

Background

References 9 through 17 transmitted Technical Specification changes necessary for the transition to ATRIUM-9B fuel at Quad Cities, Dresden, and LaSalle Nuclear Power Stations. Nuclear Regulatory Commission Safety Evaluation Reports (NRC SERs) have been issued for these Technical Specification changes. This letter transmits additional proposed revisions to the Technical Specifications for all three ComEd BWRs. These revisions are necessary to implement additional reloads of Siemens Power Corporation (SPC) ATRIUM-9B fuel and to fully utilize the applicable SPC generic methodologies. Since ComEd's initiation of the Reference 9 through 17 submittals, SPC has submitted new topical reports to the NRC for review (References 1 and 7), the NRC has approved the SPC generic topical report on ANFB application to coresident fuel (Reference 3), and various items were identified during the reload processes for LaSalle Unit 2 Cycle 8, Dresden Unit 3 Cycle 15, and Quad Cities Unit 2 Cycle 15 that need to be revised in the Technical Specifications to facilitate future reloads at these sites.

A001

This Technical Specification amendment proposes to insert methodologies that have not yet received NRC approval, i.e. ANF-91-048(P) Supplement 1, "BWR Jet Pump Model Revision for RELAX" and ANF-1125(P), Supplement 1 Appendix D, "ANFB Critical Power Correlation Uncertainty For Limited Data Sets" (References 1 and 7). NRC approval of these methodologies is required prior to approval of this amendment. If ComEd is not able to implement the Reference 1 revised jet pump model methodology, LaSalle Unit 2 Cycle 8 will experience a mid cycle derate. Additionally, if ComEd is not able to implement the Reference 7 revised ATRIUM-9B additive constant uncertainty methodology, ComEd will have to continue to calculate MCPR Safety Limits using an interim conservative ATRIUM-9B additive constant uncertainty that is not based upon a generic methodology and could eventually limit plant operation. These topical reports should be listed in Section 6 of the appropriate station's Technical Specifications as an "(A)" version. Therefore, SPC will reissue an "(A)" version of References 1 and 7 following NRC approval.

Schedule

ComEd is requesting that this application for amendment be reviewed and approved by the NRC Staff prior to February 15, 1998. Approval by February 15, 1998 will adequately support startup of the first cycle, Dresden Unit 2 Cycle 16, requiring the revisions proposed in this amendment.

Dresden will implement this amendment prior to startup of Dresden Unit 2 Cycle 16, which is currently scheduled to startup mid April 1998. Quad Cities will implement this amendment prior to startup of Quad Cities Unit 1 Cycle 16, which is currently scheduled to startup mid October 1998. LaSalle Unit 2 will implement this Technical Specification prior to startup of LaSalle Unit 2 Cycle 8, which is currently scheduled for May 1, 1998. LaSalle Unit 1 will implement this amendment prior to startup of the first Unit 1 cycle with a reload of ATRIUM-9B, LaSalle Unit 1 Cycle 9, which is currently scheduled to startup mid December 1998.

The following outlines ComEd's proposed amendment request.

1.) Attachment A provides a description and evaluation of the proposed changes to Facility Operating Licenses DPR-29, DPR-30, DPR-19, DPR-25, NPF-11, and NPF-18.

2.) Attachment B includes a summary of the proposed changes.

3.) Attachment C provides the marked up pages for Quad Cities Technical Specifications.

4.) Attachment D provides the marked up pages for Dresden Technical Specifications.

5.) Attachment E provides the marked up pages for LaSalle Unit 1 Technical Specifications.

6.) Attachment F provides the marked up pages for LaSalle Unit 2 Technical Specifications.

7.) Attachment G describes ComEd's evaluation performed in accordance with 10CFR50.90, confirming that no significant hazard consideration is involved.

8.) Attachment H provides the Environmental Assessment Applicability Review.

9.) Attachment I lists the references used in this Technical Specification amendment.

This request for amendment has been reviewed and approved by ComEd On-Site and Off-Site Review in accordance with ComEd procedures.

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.

If you have any questions concerning this letter, please contact this office.

Respectfully,

John Hosmer

John Hosmer
Engineering Vice President

cc: A. Beach, NRC Region III Administrator
NRC Senior Resident Inspector - Quad Cities
NRC Senior Resident Inspector - Dresden
NRC Senior Resident Inspector - LaSalle
R. Pulsifer, Project Manager - NRR - Quad Cities
J.F. Stang, Project Manager, NRR - Dresden
D.M. Skay, Project Manager - NRR - LaSalle
Office of Nuclear Facility Safety – IDNS
Chron–DG97-001117

STATE OF ILLINOIS

COUNTY OF DUPAGE

Docket Nos. 50-254
50-265
50-237
50-249
50-373
50-374

IN THE MATTER OF

COMMONWEALTH EDISON COMPANY

QUAD CITIES STATION - UNITS 1 & 2
DRESDEN STATION - UNITS 2 & 3
LASALLE STATION - UNITS 1 & 2

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

John Hosmer
John Hosmer
Engineering Vice President

Subscribed and sworn to before me, a Notary Public in and for the State and County above named, this 29th day of August, 1997. My commission expires on Dec. 15, 1997.

Jacqueline T. Evans
Notary Public

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

List of Attachments

A. Description and Evaluation of Proposed Changes

B. Summary of Proposed Changes

C. Marked Up Pages and Inserts for Quad Cities Technical Specifications

D. Marked Up Pages and Inserts for Dresden Technical Specifications

E. Marked Up Pages and Inserts for LaSalle Unit 1 Technical Specifications

F. Marked Up Pages and Inserts for LaSalle Unit 2 Technical Specifications

G. Evaluation of Significant Hazards Considerations

H. Environmental Assessment Applicability Review

I. References

50-237 CEC DRESDEN 2

PROPOSED CHANGE TO TECH SPECS FOR
TRANSITION TO SIEMENS POWER CORP
ATRIUM-9B FUEL.

REC'D W?LTR DTD 08/29/97....9709090102

Attachment A

Description and Evaluation of Proposed Changes

ATTACHMENT A

DESCRIPTION AND EVALUATION OF PROPOSED CHANGES

Table of Contents

ATTACHMENT A

DESCRIPTION AND EVALUATION OF PROPOSED CHANGES

1. BACKGROUND INFORMATION

This Technical Specification amendment proposes to make various changes that can be categorized into six different topics, which are listed below. Some of these topics are applicable to both operating units at Dresden, Quad Cities, and LaSalle, and others are only applicable to specific stations. The six different topics are:

1. Addition of SPC Revised Jet Pump Methodology (LaSalle Units 1 and 2)
2. Addition of SPC Generic Methodology for Application of ANFB Critical Power Correlation to Non-SPC Fuel (Quad Cities Units 1 and 2 and LaSalle Units 1 and 2)
3. Addition of SPC Topical for Revised ANFB Correlation Uncertainty (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)
4. Change to Minimum Critical Power Ratio Safety Limit (Quad Cities Units 1 and 2 and Dresden Units 2 and 3)
5. Removal of Footnotes Limiting Operation with ATRIUM-9B Fuel Reloads (Quad Cities Unit 2 and Dresden Unit 3)
6. Revision to Thermal Limit Descriptions (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

The proposed Technical Specification revision item 1, "Addition of SPC Revised Jet Pump Methodology", is applicable only to LaSalle Units 1 and 2. This amendment proposes to add **Siemens Power Corporation's (SPC) latest revision of their Loss of Coolant Accident Emergency Core Cooling System (LOCA ECCS) methodology, the revised jet pump model (Reference 1), to** Bases Section 3/4.2 and Section 6 of the LaSalle Technical Specifications. This document has not received Nuclear Regulatory Commission (NRC) approval, but is currently in NRC review.

The proposed Technical Specification revision item 2, "Addition of SPC Generic Methodology for Application of ANFB Critical Power Correlation to Non-SPC Fuel" is applicable to Quad Cities Units 1 and 2 and LaSalle Units 1 and 2. This methodology describes how the critical power ratio (CPR) calculations are performed for the coresident GE fuel, using the ANFB Critical Power Correlation. This revision proposes to add SPC's generic methodology to Bases Section 2.1.2 and Section 6 of the LaSalle Technical Specifications and Section 6 of the Quad Cities Technical Specifications. This methodology has received NRC review and approval (Reference 3).

The proposed Technical Specification revision item 3, "Addition of SPC Topical for Revised ANFB Correlation Uncertainty" is applicable to Quad Cities Units 1 and 2, LaSalle Units 1 and 2, and Dresden Units 2 and 3. This revision proposes to add SPC's methodology used to determine the ATRIUM-9B additive constant uncertainty (Reference 7) to Bases Section 2.1.2 (LaSalle) and Section 6. This methodology has not yet received NRC approval, but has been submitted to the NRC for review.

The proposed Technical Specification revision item 4, "Change to Minimum Critical Power Ratio Safety Limit" proposes to change the Minimum Critical Power Ratio (MCPR) Safety Limits of Quad Cities Units 1 and 2 to MCPR Safety Limits that will be applicable for future cycles of ATRIUM-9B reload fuel using the References 3, 6, and 7 methodologies. The MCPR Safety Limit for Dresden Units 2 and 3 is also being revised to be applicable for future cycles of ATRIUM-9B reload fuel using References 6 and 7 methodologies. References 3 and 6 are NRC approved documents, however, Reference 7, which determines the additive constant uncertainty for the ATRIUM-9B fuel, has not been approved by the NRC.

The proposed Technical Specification revision item 5, "Removal of Footnotes Limiting Operation with ATRIUM-9B Fuel Reloads", deletes footnotes that were added to the Quad Cities and Dresden Technical Specifications prior to startup of Quad Cities Unit 2 Cycle 15 and Dresden Unit 3 Cycle 15. These footnotes were added due to concerns regarding the database used by SPC for calculating the ATRIUM-9B additive constant uncertainty, as well as concerns regarding the use of a cycle specific application of ANFB to calculate the CPR of the SPC and coresident GE fuel for Quad Cities Unit 2 Cycle 15. Removal of these footnotes also allows the removal of the Quad Cities Unit 2 specific "a" pages, 2-1a and B2-3a, in the Quad Cities Technical Specifications.

Finally, the proposed Technical Specification change item 6, "Revision to Thermal Limit Descriptions", changes the description of the APLHGR Technical Specification in Section 3 for Dresden, Quad Cities, and LaSalle to state that the APLHGR limits are specified in the COLR, which is consistent with NUREG 1433/1434 (Improved Technical Specifications). Additionally, Dresden Technical Specification 3.11.D is changed to state that the linear heat generation rate (LHGR) shall not exceed the Steady State LHGR (SLHGR) limits specified in the COLR. This change is also consistent with NUREG 1433/1434.

This submittal is written assuming that References 1 and 7 receive NRC approval as they were submitted on May 6, 1996 and April 18, 1997, respectively. This submittal proposes to insert Reference 1 into Section 6 of the LaSalle Technical Specifications and Reference 7 into Section 6 of the Technical Specifications of Quad Cities, Dresden, and LaSalle.

Addition of SPC Revised Jet Pump Methodology (LaSalle Units 1 and 2)

ComEd's LOCA ECCS analysis calculations are performed by SPC for Siemens fuel. SPC has submitted to the NRC a revision (Reference 1) to the BWR jet pump model (Reference 2) to revise their ECCS evaluation methodology (Reference 8). This methodology will be used for LOCA ECCS calculations to determine the Peak Cladding Temperature (PCT) and the

Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) values. The revised LOCA ECCS methodology would first be used to prevent a mid-cycle derate during operation of LaSalle Unit 2 Cycle 8, which is currently scheduled to startup May 1, 1998. The revised LOCA ECCS jet pump model is not being inserted into Section 6 of the Quad Cities or Dresden Technical Specifications at this time because Dresden and Quad Cities Stations have different jet pump designs which causes their LOCA ECCS evaluation to be less affected by the revision to the jet pump model. The PCTs at Dresden and Quad Cities, will continue to be conservative without utilizing the Reference 1 methodology.

The revised SPC jet pump model LOCA ECCS methodology is described in the Reference 1 Topical Report. This revised jet pump model LOCA methodology was submitted on May 6, 1996 to the NRC for review. Following NRC approval, this amendment proposes to reference this methodology in the LaSalle Technical Specifications. Consistent with NRC Generic Letter 88-16, the approved Topical Reports which describe the methodologies used to determine core operating limits are to be referenced in the Technical Specifications. Bases Section 3/4.2 also references the documents that describe the calculational models used to perform the LOCA analysis. Therefore, Bases Section 3/4.2 and Section 6.6.A.6.b of the LaSalle Unit 1 and 2 Technical Specifications are being revised to reference the following Topical Report: ANF-91-048(P) Supplement 1, "BWR Jet Pump Model Revision for RELAX". (Upon NRC approval and reissue by SPC, it is expected that the NRC will include the "(A)" nomenclature in the actual Technical Specification revised pages for these references.)

Addition of SPC Generic Methodology for Application of ANFB Critical Power Correlation to Non-SPC Fuel (Quad Cities Units 1 and 2 and LaSalle Units 1 and 2)

Currently LaSalle Station and Quad Cities Station are undergoing a transition from General Electric (GE) to SPC fuel including associated methodologies. LaSalle Unit 2 Cycle 8 and Quad Cities Unit 2 Cycle 15 are the first cycles at these two plants to load SPC ATRIUM-9B fuel. This transition will be continued with reloads of SPC ATRIUM-9B fuel for LaSalle Unit 1 Cycle 9 and Quad Cities Unit 1 Cycle 16.

Due to the transition to SPC fuel at LaSalle and Quad Cities it was necessary for SPC to provide a methodology for application of their ANFB critical power correlation to the coresident GE fuel. Pending the approval of a generic ANFB application to coresident fuel, SPC produced cycle specific methodologies for the transition cores at Quad Cities and LaSalle Stations (Reference 5 for Quad Cities Unit 2 Cycle 15 and Reference 4 for LaSalle Unit 2 Cycle 8). On May 9, 1997 the NRC approved the SPC generic methodology for ANFB application to coresident fuel (Reference 3).

The SPC generic ANFB application to coresident fuel is described in Reference 3. Consistent with the NRC Generic Letter 88-16, the approved Topical Reports which describe the methodologies used to determine core operating limits are to be referenced in the Technical

Specifications. LaSalle Bases Section 2.1.2 references the documents containing "The bases for the fuel-related uncertainties". The methodology in Reference 3 will be used to determine one of the inputs to the MCPR Safety Limit calculation. Therefore, the Quad Cities Section 6 and LaSalle Technical Specifications Bases Section 2.1.2 and Section 6 are being revised to reference the following Topical Report: EMF-1125(P)(A), Supplement 1 Appendix C, "ANFB Critical Power Correlation Application for Coresident Fuel", August 1997.

Addition of SPC Topical for Revised ANFB Correlation Uncertainty (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

Due to conclusions from a recent NRC vendor performance inspection review, SPC determined that the need existed to increase the size of the critical power data base for determining the additive constant uncertainty of SPC's 9x9 fuel designs with an internal water channel (ATRIUM-9B). SPC calculated a new additive constant uncertainty for the ATRIUM-9B fuel by including additional experimental data from critical power tests of other fuel designs which share many of the same design features as the ATRIUM-9B design. The size of the database increased from 125 points to 527 points. Because the additional data included fuel designs with slight variations in the spacer design, the ATRIUM-9B additive constant uncertainty increased. Also, the additional data allowed information to be selected which addressed the full operating range of the fuel. Reference 7 presents the statistical analysis performed on the data. This document was submitted to the NRC for review and approval on April 18, 1997.

Currently, an interim conservative ATRIUM-9B additive constant uncertainty is being used to calculate the MCPR Safety Limits for Quad Cities Unit 2 Cycle 15, LaSalle Unit 2 Cycle 8, and Dresden Unit 3 Cycle 15. The conservative ATRIUM-9B additive constant uncertainty was determined by calculating the difference between the ATRIUM-9B additive constant uncertainties prior to and after the data set was expanded from 125 to 527 points. This difference was then doubled and added to the original ATRIUM-9B additive constant uncertainty. The resulting value, 0.029, is being used as the ATRIUM-9B additive constant uncertainty until NRC approval of the 0.0195 additive constant uncertainty documented in Reference 7.

The interim ATRIUM-9B additive constant uncertainty was implemented for ComEd plants in the second quarter of 1997 in order to give the NRC ample time to conduct a thorough review of the generic methodology transmitted by SPC.

This amendment proposes to add "ANF-1125(P), Supplement 1, Appendix D, "ANFB Critical Power Correlation Uncertainty for Limited Data Sets" into Section 6.9.A.6.b of the Quad Cities and Dresden Technical Specifications and Bases Section 2.1.2 and Section 6.6.A.6.b of the LaSalle Units 1&2 Technical Specifications. (Upon NRC approval and reissue by SPC, it is expected that the NRC will include the "(A)" nomenclature in the actual Technical Specification revised pages for these references.) NRC approval of Reference 7 and its incorporation into Bases Section 2.1.2 (LaSalle) and Section 6 will allow the use of an ATRIUM-9B additive constant uncertainty of 0.0195 for the MCPR Safety Limit calculations. LaSalle Bases Section 2.1.2 references the documents containing "The bases for the fuel-related uncertainties". Adding Reference 7 to Section 6 for the three ComEd BWRs is consistent with the NRC Generic Letter 88-16, which states that the approved Topical Reports which describe the methodologies used to determine core operating limits are to be referenced in the Technical Specifications.

Change to Minimum Critical Power Ratio Safety Limit (Quad Cities Units 1 and 2 and Dresden Units 2 and 3)

SPC has evaluated the MCPR Safety Limit for Quad Cities using the Reference 3 NRC approved methodology and the Reference 7 methodology (currently under NRC review). The significance of using the Reference 7 methodology is using an ATRIUM-9B additive constant uncertainty of 0.0195 in the MCPR Safety Limit calculations.

Quad Cities Unit 2 currently has a MCPR Safety Limit of 1.10, calculated using the interim conservative ATRIUM-9B additive constant uncertainty of 0.029. When the ATRIUM-9B additive constant uncertainty is decreased to 0.0195, a MCPR Safety Limit of 1.09 is supportable. For two-loop operation and an ATRIUM-9B additive constant uncertainty of 0.020 (rounded up from 0.0195), a MCPR Safety Limit of 1.09 is supported with 0.0737% rods in boiling transition (which is less than the 0.10% limit). Quad Cities Unit 1 is currently an all GE core and has a MCPR Safety Limit of 1.07. This is based on GE methodology because Unit 1 currently does not contain SPC fuel (Cycle 16 will be the first reload of ATRIUM-9B for Quad Cities Unit 1). Using the SPC ANFB critical power correlation methodology and the 0.0195 ATRIUM-9B additive constant uncertainty from Reference 7, it is estimated that the MCPR Safety Limit for Unit 1 for future SPC reloads would need to be increased to 1.09. SPC will confirm the applicability of this MCPR Safety Limit on a cycle by cycle basis. Therefore, this Technical Specification amendment proposes to revise the Quad Cities Units 1 and 2 MCPR Safety Limit to 1.09, based on Reference 7 methodology. This amendment also proposes to remove the paragraph in Bases Section 2.1.B of the Quad Cities Technical Specifications that discusses the methodology for each unit's MCPR Safety Limit calculation. The value of 1.09 is anticipated to bound actual MCPR Safety Limit calculations that would be performed for Quad Cities with future SPC reloads. ComEd would submit a future Technical Specification if actual MCPR Safety Limits are determined to be greater than 1.09.

SPC has evaluated the MCPR Safety Limit for Dresden using the Reference 6 NRC approved methodology and the Reference 7 methodology (currently in NRC review), which establishes the ATRIUM-9B additive constant uncertainty of 0.0195. Due to the differences in cycle to cycle core designs, this Technical Specification revision proposes to increase the MCPR Safety Limit for both Dresden Unit 2 and 3 from 1.08 to a more bounding value of 1.09. For Dresden Unit 3 Cycle 15, the ATRIUM-9B additive constant uncertainty of 0.029 supported a MCPR Safety Limit of 1.08 for two loop operation with 0.0997% of the fuel rods in boiling transition, and a MCPR Safety Limit of 1.09 for single loop operation, with 0.0652% of the fuel rods in boiling transition. Also for Dresden Unit 3 Cycle 15, an additive constant uncertainty of 0.020 (rounded up from 0.0195) was evaluated for two loop operation and found that 0.0405% of the fuel rods were in boiling transition at a MCPR Safety Limit of 1.08. Increasing the MCPR Safety Limit to 1.09 will result in less than 0.0405% of the fuel rods in the core in boiling transition for an additive constant uncertainty of 0.0195, and is bounding relative to a value of 1.08. The MCPR Safety Limit is being increased to bound future cycle core design differences. Cycle specific MCPR calculations will be performed for future reloads, consistent with SPC approved methodology, to confirm the continued applicability of a 1.09 MCPR Safety Limit. ComEd would submit a future Technical Specification if actual MCPR Safety Limits are determined to be greater than 1.09.

LaSalle's MCPR Safety Limits remain unchanged. The current MCPR Safety Limit evaluations for LaSalle Unit 2 Cycle 8, using the ANFB critical power correlation with an interim ATRIUM-9B additive constant uncertainty of 0.029, show that the MCPR safety limit value of 1.07 is still supported. The evaluation of the MCPR Safety Limit for LaSalle using the ATRIUM-9B additive constant uncertainty of 0.0195 from Reference 7 also yields a MCPR Safety Limit below 1.07. Therefore, the LaSalle MCPR Safety Limits remain unchanged. However, cycle specific MCPR calculations will be performed for future reloads, consistent with SPC approved methodology, to verify applicability of the 1.07 MCPR Safety Limit. ComEd would submit a future Technical Specification if actual MCPR Safety Limits are determined to be greater than 1.07.

Removal of Footnotes Limiting Operation with ATRIUM-9B Fuel Reloads (Quad Cities Unit 2 and Dresden Unit 3)

Quad Cities Technical Specifications contain footnotes in Section 2.1.B and Bases Section 2.1.B that state that the Unit 2 MCPR Safety Limit is applicable to Unit 2 Cycle 15 only. Section 6.9.A.6.b of the Quad Cities Technical Specifications contains a footnote that clarifies the cycle-specific ANFB critical power ratio correlation application to coresident fuel as being applicable only to Quad Cities Unit 2 Cycle 15. Section 5.3 of Dresden's Technical Specifications contains footnotes that allow operation with ATRIUM-9B reloads in all modes for Dresden Unit 3, Cycle 15, only. Another footnote limits the use of ATRIUM-9B fuel in Unit 2, with the exception of lead test assemblies, to Operational Modes 3, 4, and 5, and with no more than one control rod withdrawn.

The footnotes in the Quad Cities Technical Specifications were added because the MCPR Safety Limit calculated for Quad Cities Unit 2 Cycle 15 was based on a cycle specific methodology (Reference 5), and therefore, is only applicable to Quad Cities Unit 2 Cycle 15. The footnotes in the Dresden Technical Specifications were added due to concerns regarding the SPC methodology used to determine the ATRIUM-9B additive constant uncertainty for MCPR Safety Limit calculations.

Reference 3 is the SPC generic methodology for applying the ANFB critical power ratio correlation to non-SPC fuel. The Reference 3 methodology has been approved by the NRC and is proposed to be inserted into the Quad Cities and LaSalle Technical Specifications in this submittal. This submittal also proposes to insert the SPC topical addressing the ATRIUM-9B additive constant uncertainties, Reference 7, into all three ComEd BWR Technical Specifications following NRC approval of the document. Because these two documents generically address the concerns associated with the MCPR Safety Limit methodology, the footnotes can be removed.

Removal of these footnotes also allows the removal of the Unit 2 specific "a" pages, 2-1a and B2-3a, in the Quad Cities Technical Specifications.

Revision to Thermal Limit Descriptions (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

The Average Planar Linear Heat Generation Rate (APLHGR) limit is established to ensure that the Peak Clad Temperature (PCT) and maximum oxidation limits specified in 10 CFR 50.46 will not be exceeded. The 10 CFR 50.46 limits are maintained by operating within the Maximum APLHGR (MAPLHGR) limits of the Core Operating Limits Report (COLR). SPC currently calculates the APLHGR limits based on a LOCA analysis and uses models consistent with the requirements of Appendix K of 10 CFR Part 50.

Currently the Dresden Technical Specifications stipulate that the APLHGR limit be a function of bundle average exposure and both Quad Cities and LaSalle Technical Specifications stipulate that the APLHGR limit be a function of average planar exposure. The purpose of this transmittal is to request generalization of the definition of the APLHGR limits to allow either bundle average or average planar exposure based APLHGR limits, consistent with LOCA analyses of record. This generalization of the definition of APLHGR is consistent with NUREG 1433/1434 (Improved Technical Specification) wording. Both MAPLHGRs (bundle average exposure based and planar average exposure based) are acceptable for Appendix K of 10 CFR Part 50. The generalization of the APLHGR would allow the COLR exclusively to identify the APLHGR limits and their exposure basis.

The LHGR is limited to ensure that fuel integrity limits are not exceeded. Currently the Dresden Technical Specifications stipulate that the LHGR limit be a function of average planar exposure. This transmittal requests generalization of the definition of the LHGR limit, which is consistent with NUREG 1433/1434 wording. The generalization of the LHGR would allow the COLR exclusively to identify the LHGR limits and their exposure basis.

2. DESCRIPTION OF THE PROPOSED CHANGES

Addition of SPC Revised Jet Pump Methodology (LaSalle Units 1 and 2)

The proposed change adds the Reference 1 revised LOCA methodology to Bases Section 3/4.2 and Section 6.6.A.6.b of the LaSalle Units 1 and 2 Technical Specifications. SPC has submitted a revision (Reference 1) to the BWR jet pump model (Reference 2) to the NRC which revises their ECCS evaluation methodology (Reference 8). The revised jet pump model changes the calculational behavior in the jet pump under reversed drive flow conditions. Therefore, the revised jet pump model exhibits a more realistic behavior and produces small break LOCA PCTs that are comparable to the large break LOCA results. The Reference 1 revised jet pump model will continue to ensure fuel design criteria and 10CFR50.46 compliance.

Addition of SPC Generic Methodology for Application of ANFB Critical Power Correlation to Non-SPC Fuel (Quad Cities Units 1 and 2 and LaSalle Units 1 and 2)

The SPC critical power correlation, ANFB, will be the CPR correlation of record for both the GE and SPC coresident fuel at Quad Cities and LaSalle, and will be used to determine the MCPR operating limits resulting from analyses of abnormal operational occurrences. The MCPR of the coresident GE fuel will be calculated using bundle geometry dependent additive constants determined as described in Reference 3 with the ANFB calculated MCPR being conservative relative to the MCPR calculated by the GE correlation (GEXL).

All references to cycle specific topicals relating to the use of the ANFB CPR correlation for GE fuel from both LaSalle Unit 2 (Reference 4) and Quad Cities Unit 2 (Reference 5) Technical Specifications will be removed and the approved generic ANFB application to coresident fuel topical (Reference 3) will be added. This topical is not only applicable for the current cycles which contain coresident GE fuel but also for all future cycles that utilize irradiated GE fuel and fresh/irradiated SPC fuel in the core at Quad Cities and LaSalle stations.

Addition of SPC Topical for Revised ANFB Correlation Uncertainty (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

This submittal proposes to include the SPC analysis of the expanded data set for calculating the ATRIUM-9B additive constant uncertainty to Section 6.9.A.6.b of the Technical Specifications of Quad Cities and Dresden and Bases Section 2.1.2 and Section 6.6.A.6.b of the LaSalle Units 1 and 2 Technical Specifications. This SPC topical report (Reference 7) describes the methodology used to justify the ATRIUM-9B additive constant uncertainties for the ANFB Critical Power Correlation. Since Section 6.9.A.6.b of the Dresden and Quad Cities Technical

Specifications and Section 6.6.A.6.b of the LaSalle Technical Specifications provide the "analytical methods used to determine the operating limits" as stated in NRC Generic Letter 88-16, it is appropriate that Reference 7 be included. Because, LaSalle's Bases Section 2.1.2 references the documents containing "The bases for the fuel-related uncertainties", it is appropriate that Reference 7 be added to LaSalle's Bases Section 2.1.2.

Change to Minimum Critical Power Ratio Safety Limit (Quad Cities Units 1 and 2 and Dresden Units 2 and 3)

It is envisioned that the ATRIUM-9B additive constant uncertainty used to calculate the MCPR Safety Limit for future cycles will be 0.0195 as documented in Reference 7. Reference 7 describes how SPC determined this ATRIUM-9B additive constant uncertainty for an expanded critical power data set of 9x9 fuel with an internal water box.

This Technical Specification amendment proposes to change the MCPR Safety Limit for Quad Cities Units 1 and 2 and Dresden Units 2 and 3. The MCPR Safety Limit for Quad Cities Unit 1 will be increased from 1.07 to 1.09. The MCPR Safety Limit for Quad Cities Unit 2 will be decreased from 1.10 to 1.09. Additionally, the paragraph that explains that Unit 1's MCPR Safety Limit methodology is based on GEXL and that Unit 2's MCPR Safety Limit methodology is based on ANFB critical power correlation in Bases Section 2.1.B will no longer be necessary and will be removed.

The Dresden Units 2 and 3 MCPR Safety Limit is currently 1.08 and is proposed to be increased to 1.09. The MCPR Safety Limit is increased for Dresden to bound operation with ATRIUM-9B fuel for future cycles.

Removal of Footnotes Limiting Operation with ATRIUM-9B Fuel Reloads (Quad Cities Unit 2 and Dresden Unit 3)

Quad Cities Technical Specifications contain footnotes in Section 2.1.B and Bases Section 2.1.B that clarify that the Unit 2 Safety Limit MCPR is applicable to Unit 2 Cycle 15 only. Section 6.9.A.6.b of the Quad Cities Technical Specifications contains a footnote that clarifies the cycle-specific application of ANFB critical power correlation to coresident fuel as being applicable only to Quad Cities Unit 2 Cycle 15. Section 5.3 of Dresden's Technical Specifications also contains footnotes that allow operation with ATRIUM-9B fuel in all modes for Dresden Unit 3, Cycle 15, only. Additional footnotes in Section 5.3 also limit the use of ATRIUM-9B fuel in Unit 2, with the exception of lead test assemblies, to Operational Modes 3, 4, and 5, and with no more than one control rod withdrawn and state that the design bases are applicable in Operational Modes 3, 4, and 5 for Unit 2 only. The proposed change is to delete all of these footnotes.

Removal of these footnotes also allows the removal of the Unit 2 specific "a" pages, 2-1a and B2-3a, in the Quad Cities Technical Specifications.

Revision to Thermal Limit Descriptions (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

Dresden Technical Specification 3.11.A requires that all APLHGRs for each type of fuel be a function of bundle average exposure and not exceed the limits specified in the Core Operating Limits Report (COLR). The proposed change is to remove the words "for each type of fuel as a function of bundle average exposure" in order to maintain consistency with calculated results for the determination of the APLHGR and maintain compliance with Appendix K of 10 CFR Part 50 for the calculation of the APLHGR. Quad Cities Specification 3.11.A and LaSalle Specification 3.2.1 require that all APLHGRs for every fuel type be a function of average planar exposure. The proposed change is to remove "for each type of fuel as a function of average planar exposure" in order to establish consistent wording among the sites and maintain compliance with Appendix K of 10 CFR Part 50 for the calculation of the APLHGR. Thus the new description for Quad Cities, Dresden, and LaSalle would state, "All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGR) shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT." This change is consistent with NUREG 1433/1434. The definition of Average Planar Exposure is also deleted from the Definitions section of the Technical Specifications for all three BWRs.

Currently the Dresden Technical Specifications stipulate that the LHGR limit be a function of average planar exposure. The proposed change is to remove the words "for each type of fuel as a function of AVERAGE PLANAR EXPOSURE". Thus the new description of LHGR for Dresden would state "The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed the STEADY STATE LINEAR HEAT GENERATION RATE (SLHGR) limits specified in the CORE OPERATING LIMITS REPORT (COLR). This change is consistent with NUREG 1433/1434 wording. The generalization of the LHGR would allow the COLR exclusively to identify the LHGR limits and their exposure basis.

3. DESCRIPTION OF THE CURRENT REQUIREMENTS

Addition of SPC Revised Jet Pump Methodology (LaSalle Units 1 and 2)

Reference 8 describes the NRC-approved methodology currently used by SPC for LaSalle LOCA analyses. The methodology in Reference 8 is used to ensure compliance with fuel design criteria and 10CFR50.46 requirements. Currently the jet pump model is described in Reference 2. Applications of RELAX for BWR LOCAs (particularly small break LOCAs with breaks in the recirculation loop pump discharge piping) have calculated unrealistic behavior in the jet pump under reversed drive flow conditions. A result of this unrealistic jet pump model is overly conservative PCTs from the LOCA analysis, resulting in overly conservative APLHGR limits. These overly conservative APLHGR limits will limit operation of LaSalle Unit 2 Cycle 8 mid-cycle to less than rated power.

Addition of SPC Generic Methodology for Application of ANFB Critical Power Correlation to Non-SPC Fuel (Quad Cities Units 1 and 2 and LaSalle Units 1 and 2)

Currently, the determination of MCPR for GE coresident fuel in Quad Cities Unit 2 Cycle 15 and LaSalle Unit 2 Cycle 8 is based on the SPC ANFB critical power correlation and is prescribed per cycle specific documentation. This cycle specific documentation is referenced in section 6.9.A.6.b of the Quad Cities Technical Specifications as Reference 19, *ComEd letter, "ComEd Response to NRC Staff Request for Additional Information (RAI) Regarding the Application of Siemens Power Corporation ANFB Critical Power Correlation to Coresident General Electric Fuel for LaSalle Unit 2 Cycle 8 and Quad Cities Unit 2 Cycle 15, NRC Docket No.'s 50-373/374 and 50-254/265", J.B. Hosmer to U.S. NRC, July 2, 1996, transmitting the topical report, Application of the ANFB Critical Power Correlation to Coresident GE Fuel for Quad Cities Unit 2 Cycle 15, EMF-96-051(P), Siemens Power Corporation - Nuclear Division, May 1996, and related information.* The methodology for determining the MCPR of the coresident GE fuel at LaSalle is referenced in the Bases Section 2.1.2 of the Technical Specifications for Unit 2 as Reference 6: *"Application of the ANFB Critical Power Correlation to Coresident GE Fuel for LaSalle Unit 2 Cycle 8," EMF-96-021(P), Revision 1, Siemens Power Corporation, February 1996; NRC SER letter dated September 26, 1996.*

Addition of SPC Topical for Revised ANFB Correlation Uncertainty (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

The current MCPR Safety Limit calculations for cycles containing ATRIUM-9B reloads (Quad Cities Unit 2 Cycle 15, LaSalle Unit 2 Cycle 8, and Dresden Unit 3 Cycle 15) use an interim ATRIUM-9B additive constant uncertainty of 0.029. These calculations result in a MCPR safety limit value of 1.10 for Quad Cities Unit 2, 1.07 for LaSalle Unit 2, and 1.08 for Dresden Unit 3.

The Administrative Control section of the Technical Specifications lists the NRC approved topical reports describing the analytical methods used to determine the operating limits (Specification 6.9.A.6.b for Quad Cities and Dresden and 6.6.A.6.b for LaSalle). Bases Section 2.1.2 of LaSalle's Technical Specifications references the documents containing "The bases for the fuel-related uncertainties". The methods listed include the ANFB Critical Power Correlation topical (Reference 6). Consistent with NRC Generic Letter 88-16, the approved Topical Reports which describe the methodologies used to determine core operating limits are to be referenced in the Technical Specifications.

Change to Minimum Critical Power Ratio Safety Limit (Quad Cities Units 1 and 2 and Dresden Units 2 and 3)

The current requirements for Quad Cities Unit 1 (all GE core) are based on GE methods utilizing the GE CPR correlation (GEXL) to calculate the CPR for the GE fuel bundles and the MCPR Safety Limit. The MCPR Safety Limit at Quad Cities Unit 1 is 1.07 based on GEXL results. The current requirements for Quad Cities Unit 2, which contains a reload of SPC ATRIUM-9B fuel, are based on SPC methods, including the use of the SPC ANFB critical power correlation to calculate the CPR for both SPC ATRIUM-9B fuel and GE fuel. This methodology is addressed on a cycle specific basis in Reference 5. The MCPR Safety Limit for Unit 2 is 1.10. The reason for the higher MCPR Safety Limit on Unit 2 is because the Unit 2 MCPR Safety Limit was calculated using a conservative interim additive constant uncertainty of 0.029 for the ATRIUM-9B fuel in Quad Cities Unit 2 Cycle 15, and a conservative additive constant uncertainty of 0.038 for the coresident GE fuel.

The current requirements for Dresden Units 2 and 3 are based on SPC methodology utilizing the SPC ANFB CPR correlation to calculate the CPR for the ATRIUM-9B fuel and the 9x9-2 fuel. The MCPR Safety Limit for Dresden Units 2 and 3 is 1.08. For Dresden Unit 3 Cycle 15, which is operating with a reload of ATRIUM-9B fuel, the 1.08 MCPR Safety Limit is supported even with an interim ATRIUM-9B additive constant uncertainty of 0.029.

The current requirement for LaSalle Units 1 and 2 is a MCPR Safety Limit of 1.07 for both units. The Unit 1 (all GE core) MCPR Safety Limit is based on GE methodology. The Unit 2 MCPR Safety Limit is based on the SPC ANFB critical power correlation methodology with application of the SPC ANFB correlation to GE fuel and is currently addressed in Reference 4. A MCPR Safety Limit of 1.07 can be supported for either an ATRIUM-9B additive constant uncertainty of 0.029 or 0.0195 for the current Unit 2 cycle. It is also estimated that Unit 1 can maintain a MCPR Safety Limit of 1.07 for either an ATRIUM-9B additive constant uncertainty of 0.029 or 0.0195 using SPC methodology for future cycles. However, cycle specific MCPR calculations will be performed for future reloads, consistent with SPC approved methodology, to confirm the continued applicability of the 1.07 MCPR Safety Limit. Therefore, no changes are necessary to the MCPR Safety Limits at LaSalle.

DESCRIPTION AND EVALUATION OF PROPOSED CHANGES

Removal of Footnotes Limiting Operation with ATRIUM-9B Fuel Reloads (Quad Cities Unit 2 and Dresden Unit 3)

Quad Cities Technical Specifications contain footnotes in Section 2.1.B and Bases Section 2.1.B that clarify that the Unit 2 Safety Limit MCPR is applicable to Unit 2 Cycle 15 only. Section 6.9.A.6.b of the Quad Cities Technical Specifications contains a footnote that clarifies the cycle-specific ANFB critical power correlation application to coresident fuel as being applicable only to Quad Cities Unit 2 Cycle 15. Section 5.3 of Dresden's Technical Specifications also contains footnotes that allow operation with ATRIUM-9B fuel in all modes for Dresden Unit 3, Cycle 15, only, and another footnote in Section 5.3 limiting the use of ATRIUM-9B fuel in Unit 2, with the exception of lead test assemblies, to Operational Modes 3, 4, and 5 and with no more than one control rod withdrawn. Both Dresden Unit 2 and Quad Cities Unit 1 are currently able to operate with both of these footnotes; however, because both of these units are scheduled to reload SPC's ATRIUM-9B fuel for Dresden Unit 2 Cycle 16 and Quad Cities Unit 1 Cycle 16, operation with ATRIUM-9B fuel will be prohibited with these footnotes.

Currently the Quad Cities Technical Specifications contain "a" pages, 2-1a and B2-3a, that were created specifically for Unit 2 due to these footnotes.

Revision to Thermal Limit Descriptions (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

Section 3.11.A of the Dresden Technical Specifications currently states, "All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGR) for each type of fuel as a function of bundle average exposure shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT." Section 3.11.A of the Quad Cities and Section 3.2.1 of the LaSalle Technical Specifications currently state, "All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGR) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT." This limit is applicable when in Operational Mode 1 at thermal power equal to or greater than 25% Rated Thermal Power.

Section 3.11 D of the Dresden Technical Specifications currently states, "The LINEAR HEAT GENERATION RATE (LHGR) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the STEADY STATE LINEAR HEAT GENERATION RATE (SLHGR) limits specified in the CORE OPERATING LIMITS REPORT." This limit is applicable when in Operational Mode 1 at thermal power equal to or greater than 25% Rated Thermal Power.

4. BASES FOR THE CURRENT REQUIREMENTS

Addition of SPC Revised Jet Pump Methodology (LaSalle Units 1 and 2)

Reference 8 is the NRC-approved methodology currently in place for SPC LOCA analyses. The methodology in Reference 8 is used to ensure compliance with fuel design criteria and 10CFR50.46 requirements. Currently the jet pump model is described in Reference 2. Applications of RELAX for BWR LOCAs (particularly small break LOCAs with breaks in the recirculation loop pump discharge piping) have calculated unrealistic behavior in the jet pump under reversed drive flow conditions. Reference 8 will remain in Section 6 of the Technical Specifications and will be supplemented by Reference 1.

Addition of SPC Generic Methodology for Application of ANFB Critical Power Correlation to Non-SPC Fuel (Quad Cities Units 1 and 2, LaSalle Units 1 and 2)

The current MCPR requirements are based on cycle specific SPC application of the ANFB critical power correlation to GE fuel for Quad Cities Unit 2 Cycle 15 and LaSalle Unit 2 Cycle 8 cores (Reference 5 for Quad Cities and Reference 4 for LaSalle).

The cycle specific methods are NRC approved and are used to ensure that less than 0.1% of the fuel rods are in boiling transition during anticipated operational occurrences.

The Quad Cities Unit 1 and LaSalle Unit 1 current MCPR requirements are based on GE methodology.

Addition of SPC Topical for Revised ANFB Correlation Uncertainty (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

The list of references in Section 6.9.A.6.b for Quad Cities and Dresden and Bases Section 2.1.2 and Section 6.6.A.6.b for LaSalle provides documentation in the Technical Specifications of the NRC approved methods used to determine the operating limits in the Core Operating Limits Report (COLR). This Reference list was created in response to Generic Letter 88-16.

The reference lists in Section 6 for Quad Cities and Dresden, and Bases Section 2.1.2 and Section 6 for LaSalle have been updated to include the ANFB Critical Power Correlation (Reference 6). (These pages for LaSalle Unit 1 Technical Specifications will be implemented upon startup of Unit 1 Cycle 9.)

The current requirements for the ATRIUM-9B additive constant uncertainty result from findings during a recent NRC vendor performance inspection review. SPC determined that the need existed to increase the size of the data base for determining the ATRIUM-9B additive constant uncertainty from 125 points to 527 points. SPC calculated a new additive constant uncertainty for the ATRIUM-9B fuel by including additional experimental data from critical power tests from other fuel designs which share many of the same design features as the ATRIUM-9B design. The additional experimental data was selected to address the full operating range of the fuel. Reference 7 presents the statistical analysis performed on the data and was submitted to the NRC Staff for review and approval on April 18, 1997.

Because it was anticipated that the NRC would not have sufficient time between the submittal of Reference 7 and the startups of Quad Cities Unit 2 Cycle 15 and Dresden 3 Cycle 15 to fully review Reference 7, a conservative interim ATRIUM-9B additive constant uncertainty was determined. The conservative interim additive constant uncertainty was calculated by using the difference between the ATRIUM-9B additive constant uncertainties prior to and after the data set was expanded to 527 points. This difference was doubled and added to the original additive constant uncertainty. The resulting value, 0.029, is being used as the additive constant uncertainty until NRC approval of the Reference 7 document. The following table summarizes the method used to determine the 0.029 ATRIUM-9B additive constant uncertainty.

ATRIUM-9B Additive Constant Uncertainty	**Value**
Original Additive Constant Uncertainty for ATRIUM-9B (data set of 125 points)	0.010
Revised (Reference 7) Additive Constant Uncertainty for ATRIUM-9B (data set of 527 points)	0.0195
Interim ATRIUM-9B Additive Constant Uncertainty used to calculate more conservative Q2C15/D3C15 MCPR Safety Limit = (0.010 + 2 (0.0195 - 0.010))	0.029

Change to Minimum Critical Power Ratio Safety Limit (Quad Cities Units 1 and 2 and Dresden Units 2 and 3)

The current requirements for Quad Cities Unit 1 are based on an all GE core and GE methods, including the use of the GE CPR correlation (GEXL) to calculate the MCPR Safety Limit for the GE fuel bundles. Currently, the MCPR Safety Limit at Quad Cities Unit 1 is 1.07 based on GEXL results. The current requirements for Quad Cities Unit 2, which contains a reload of ATRIUM-9B fuel, are based on SPC methods, including the use of the SPC ANFB critical power correlation to calculate the CPR for both ATRIUM-9B fuel and GE fuel. This methodology is currently addressed in Reference 5. The MCPR Safety Limit for Unit 2 is 1.10. The reason the MCPR Safety Limit is higher for Unit 2 is because the Unit 2 MCPR Safety Limit was calculated using a) the conservative interim additive constant uncertainty of 0.029 (described previously) for the ATRIUM-9B fuel in Quad Cities Unit 2 Cycle 15, and b) a conservative application of the ANFB correlation to the coresident GE fuel which results in an additive constant uncertainty of 0.038 for the coresident GE fuel. The MCPR Safety Limit was documented as cycle specific in the Bases Section 2.1.B of the Technical Specifications in order to clarify the methodology used by SPC was only applicable to Unit 2 Cycle 15.

The current requirements for Dresden Units 2 and 3 are based on SPC methods, including the use of the ANFB critical power correlation to calculate the MCPR Safety Limit. Currently, the MCPR Safety Limit for Dresden Units 2 and 3 is 1.08 based on SPC methodology (Reference 6).

Removal of Footnotes Limiting Operation with ATRIUM-9B Fuel Reloads (Quad Cities Unit 2 and Dresden Unit 3)

The footnotes in the Quad Cities Technical Specifications were added because the MCPR Safety Limit that was calculated for Quad Cities Unit 2 Cycle 15 was calculated based on a cycle specific methodology, and therefore was only applicable to Quad Cities Unit 2 Cycle 15. The footnotes in the Dresden Technical Specifications, as well as the Quad Cities Technical Specifications, were added due to concerns regarding the SPC methodology for calculating ATRIUM-9B additive constant uncertainties used in the MCPR Safety Limit calculations. The Quad Cities Technical Specifications also contain Unit 2 specific "a" pages, 2-1a and B2-3a, which were created due to these footnotes.

Revision to Thermal Limit Descriptions (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

The APLHGR limit is determined to assure that the peak clad temperature will not exceed the PCT and maximum oxidation limits defined by 10CFR50.46 following a postulated LOCA. The PCT is primarily a function of the initial condition's average heat generation rate of an assembly at any axial location in the core. Guidelines for the calculational model of the ALPHGR are provided in Appendix K of 10CFR Part 50.

The current Dresden Technical Specifications state that the APLHGR limit is a function of bundle average exposure and the current Quad Cities and LaSalle Technical Specifications state that the APLHGR limit is a function of average planar exposure. APLHGR limits can be a function of either bundle average or planar average exposure. To maintain consistency in the Technical Specification wording among the three sites a more general description is proposed.

The LHGR limit is determined to ensure that fuel integrity limits are not exceeded. The current Dresden Technical Specifications state that the LHGR limit is a function of average planar exposure. LaSalle and Quad Cities do not specify an exposure basis for the LHGR Technical

Specification. To maintain consistency in the Technical Specification wording among the three sites, a more general description is proposed.

5. NEED FOR THE REVISION OF THE REQUIREMENTS

Addition of SPC Revised Jet Pump Methodology (LaSalle Units 1 and 2)

The proposed change adds the Reference 1 revised LOCA methodology to Bases Section 3/4.2 and Section 6 of the LaSalle Unit 1 and 2 Technical Specifications. Reference 1 is a supplement to the NRC approved EXEM/BWR LOCA ECCS evaluation model (Reference 8). Applications of RELAX for BWR LOCAs (particularly small break LOCAs with breaks in the recirculation loop pump discharge piping) had calculated unrealistic behavior in LaSalle type jet pumps under reversed drive flow conditions resulting in high calculated PCT's. The revised RELAX model corrects this unrealistic calculational behavior. For a break in the recirculation pump discharge, the calculated PCT is reduced approximately 300°F using the revised method. The revised methodology is needed at LaSalle to eliminate mid-cycle derates resulting from overly conservative PCTs which lead to overly conservative APLHGR limits derived from the Reference 8 jet pump model.

Addition of SPC Generic Methodology for Application of ANFB Critical Power Correlation to Non-SPC Fuel (Quad Cities Units 1 and 2 and LaSalle Units 1 and 2)

This revision deletes the cycle specific documentation for the ANFB critical power correlation application to coresident fuel from the Technical Specifications for Quad Cities Unit 2 Cycle 15 and LaSalle Unit 2 Cycle 8 and replaces it with the NRC approved SPC generic methodology (Reference 3) in section 6 of Quad Cities and Section 6 and Bases Section 2.1.2 of LaSalle Unit 1 & 2 Technical Specifications. This change is necessary to add to the Technical Specifications a generic methodology for the application of the ANFB correlation to coresident GE fuel that is not cycle specific and that will apply for future cycles on both units at both stations.

Addition of SPC Topical for Revised ANFB Correlation Uncertainty Calculation (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

Siemens Power Corporation has modified the calculation of the ANFB additive constant uncertainties for ATRIUM-9B fuel by expanding the data base used to determine the ATRIUM-9B additive constant uncertainty. This analysis is documented in Reference 7. The ANFB methods are documented in Reference 6. Reference 7 supplements Reference 6 with the expanded data base analysis. It is appropriate that Reference 7 be added to the list of references in Section 6 of the Technical Specifications for Quad Cities and Dresden, and Bases Section 2.1.2 and Section 6 LaSalle, because Reference 7 will be the basis of the ATRIUM-9B additive constant uncertainties in the MCPR Safety Limit analysis.

Change to Minimum Critical Power Ratio Safety Limit (Quad Cities Units 1 and 2 and Dresden Units 2 and 3)

The MCPR Safety Limit for Quad Cities Unit 2 decreases from 1.10 to 1.09. The reason for changing the MCPR Safety Limit for Quad Cities Unit 2 is the revised ATRIUM-9B additive constant uncertainty, which is an input to the MCPR Safety Limit calculation. The new methodology (Reference 7) documents SPC's expansion of the critical power data set used to calculate the ATRIUM-9B additive constant uncertainty. Because the 0.0195 ATRIUM-9B additive constant uncertainty of Reference 7 is less than the 0.029 ATRIUM-9B additive constant uncertainty currently used for the conservative interim MCPR Safety Limit calculation, the MCPR Safety Limit can be decreased. This MCPR Safety Limit reduction will provide additional operational flexibility.

The MCPR Safety Limit for Quad Cities Unit 1 increases from 1.07 to 1.09. The reason the MCPR Safety Limit for Quad Cities Unit 1 will need to be revised is due to the introduction of SPC ATRIUM-9B fuel and SPC methodologies. Therefore, this Technical Specification amendment proposes to make the MCPR Safety Limit for Unit 1 consistent with SPC methodology by making it the same as the MCPR Safety Limit for Quad Cities Unit 2.

Since MCPR Safety Limits for both Quad Cities Units 1 and 2 are determined with SPC methodology, and they are both being changed to 1.09, the paragraph in Section 2.1.B clarifying that the Unit 1 MCPR Safety Limit was determined using GE methods and the Unit 2 MCPR Safety Limit was determined using SPC methods needs to be removed.

Dresden Units 2 and 3 MCPR Safety Limit is proposed to be revised from 1.08 to 1.09. This change is requested to accommodate future cycles. The results for the Dresden Unit 3 Cycle 15 MCPR Safety Limit calculation indicate that for a MCPR Safety Limit of 1.08 and an additive constant uncertainty of 0.02 (which is 0.0195 rounded up) 0.0405% of the fuel rods in the core are in boiling transition. This is well below the limit of 0.1% of the rods in the core in boiling transition. Therefore, increasing the MCPR Safety Limit to 1.09 will add additional conservatism to the number of fuel rods in boiling transition.

For future cycles, it is desirable to incorporate a MCPR Safety Limit of 1.09 into the Technical Specifications for both Dresden and Quad Cities. This value is anticipated to bound the results of future MCPR Safety Limit calculations. Should this not be the case, a future Technical Specification change would be initiated by ComEd.

Removal of Footnotes Limiting Operation with ATRIUM-9B Fuel Reloads (Quad Cities Unit 2 and Dresden Unit 3)

Because ComEd intends to use SPC's generic ANFB application to coresident fuel for calculating future cycle's MCPR Safety Limits, the footnotes in Section 2.1.B and Bases Section 2.1.B of the Quad Cities Technical Specifications need to be removed. The approval of Reference 3 supports removing these footnotes.

It is appropriate to remove the footnote in Section 6.9.A.6.b of Quad Cities Technical Specifications because this amendment proposes to remove the Quad Cities Unit 2 Cycle 15 MCPR methodology from the reference list. The Quad Cities Unit 2 Cycle 15 MCPR methodology reference will be replaced with the generic methodology reference (Reference 3) which is valid for all future cycles containing SPC and coresident GE fuel at Quad Cities.

It is appropriate to remove the footnotes in Section 5.3 of Dresden's Technical Specifications which limit operation with reloads of ATRIUM-9B to Unit 3 Cycle 15. Concerns regarding the ATRIUM-9B additive constant uncertainty used for the MCPR Safety Limit calculation will be resolved with the anticipated NRC approval of Reference 7. Currently the upcoming reloads at Dresden are planned to be SPC ATRIUM-9B reloads.

Removal of these footnotes also allows the removal of the Unit 2 specific "a" pages, 2-1a and B2-3a, in the Quad Cities Technical Specifications. Therefore, it is appropriate that this amendment requests pages 2-1a and B2-3a be removed.

Revision to Thermal Limit Descriptions (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

Currently the Dresden Technical Specifications stipulate the APLHGR limit is monitored on a bundle average exposure basis. Quad Cities and LaSalle Technical Specifications currently specify the APLHGR limit as a function of the average planar exposure basis. However, the results of SPC's NRC approved LOCA methodology may be applied on either a bundle average or average planar exposure. Thus, a less stringent description of APLHGR is proposed by this amendment such that the detailed information to which the APLHGR is monitored is specified in the COLR. The proposed revision is to remove the stipulation that APLHGR limits are to be based on either bundle average or average planar exposure. The revised wording refers the reader to the COLR for the APLHGR limits, which is consistent with NUREG 1433/1434's approved wording. Additionally, the definition of Average Planar Exposure is deleted from the Definitions section of the Technical Specifications for all three BWRs. This change would allow the most suitable method to be utilized as specified in the COLR and will establish consistency in Technical Specification wording among the sites.

DESCRIPTION AND EVALUATION OF PROPOSED CHANGES

Currently the Dresden Technical Specifications stipulate that the LHGR limit is monitored on an average planar exposure basis. Quad Cities and LaSalle Technical Specifications currently do not specify an exposure basis for the LHGR limit. Thus, a description of the LHGR limit at Dresden is proposed by this amendment such that the detailed information to which the LHGR is monitored is specified in the COLR. The proposed revision is to remove the stipulation that the LHGR limit is based on average planar exposure. The revised wording refers the reader to the COLR for the LHGR limits, which is consisted with NUREG 1433/1434's approved wording. As stated before, the definition of Average Planar Exposure is removed from Dresden's Technical Specifications. This change will allow the LHGR limit to be specified in the COLR and will establish consistency in Technical Specification wording among the sites.

6. DESCRIPTION OF THE REVISED REQUIREMENTS

Addition of SPC Revised Jet Pump Methodology (LaSalle Units 1 and 2)

The revised SPC LOCA ECCS methodology is described in the Reference 1 Topical Report. Consistent with NRC Generic Letter 88-16, the approved Topical Reports which describe the methodologies used to determine core operating limits are to be referenced in the Technical Specifications. Bases Section 3/4.2 also references the documents that describe the calculational models used to perform the LOCA analysis. Therefore, Bases Section 3/4.2 and Section 6.6.A.6.b of the LaSalle Technical Specifications are being revised to include the Reference 1 SPC LOCA ECCS Topical Report.

Addition of SPC Generic Methodology for Application of ANFB Critical Power Correlation to Non-SPC Fuel (Quad Cities Units 1 and 2 and LaSalle Units 1 and 2)

This amendment proposes to remove the cycle specific references from section 6.9.A.6.b of the Quad Cities Technical Specifications and Bases Section 2.1.2 of the LaSalle Unit 2 Technical Specifications. The generic Reference 3 methodology for ANFB critical power correlation application to coresident fuel topical is then added to section 6.9.A.6.b of the Quad Cities and Bases Section 2.1.2 and Section 6.6.A.6.b of the LaSalle Technical Specifications. Reference 3 describes the methodology used by SPC to determine the additive constant uncertainty for application of the ANFB correlation to GE coresident fuel.

Addition of SPC Topical for Revised ANFB Correlation Uncertainty (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

This amendment proposes to add Reference 7 to Bases Section 2.1.2 and Section 6.6.A.6.b of LaSalle Units 1&2, and Section 6.9.A.6.b of the Dresden and Quad Cities Technical Specifications. Reference 7 describes the process used by SPC to calculate the additive constant uncertainties for ATRIUM-9B fuel using an expanded experimental data set. Reference 7 is the basis for the calculation of a 0.0195 ATRIUM-9B additive constant uncertainty.

Change to Minimum Critical Power Ratio Safety Limit (Quad Cities Unit 1 and 2 and Dresden Units 2 and 3)

This amendment proposes to change the MCPR Safety Limits for Quad Cities and Dresden Nuclear Station. The change involves increasing the MCPR Safety Limit for Quad Cities Unit 1 from 1.07 to 1.09 and decreasing the MCPR Safety Limit for Unit 2 from 1.10 to 1.09. The MCPR Safety Limit for Dresden Units 2 and 3 is proposed to increase from 1.08 to 1.09. All

MCPR surveillance requirements remain unchanged. The change also involves removing the paragraph in Bases Section 2.1.B of the Quad Cities Technical Specifications specifying the vendor's methodology used to determine the MCPR Safety Limit.

Removal of Footnotes Limiting Operation with ATRIUM-9B Fuel Reloads (Quad Cities Unit 2 and Dresden Unit 3)

The revised requirements entail removing the following footnotes from the Quad Cities Technical Specifications: in Section 2.1.B and Bases Section 2.1.B the footnotes clarifying that the MCPR Safety Limit applies to Unit 2 Cycle 15 only and in Section 6.9.A.6.b the footnote clarifying that the cycle specific CPR ANFB application to coresident fuel applies to Unit 2 Cycle 15 only. Additionally the revised requirements involve removing the following footnotes in Section 5.3 of the Dresden Technical Specifications: the footnote permitting the use of ATRIUM-9B fuel in all modes for Unit 3 Cycle 15 only and the footnotes limiting the use of ATRIUM-9B fuel (with the exception of lead test assemblies) in Unit 2 to Modes 3, 4, and 5 with no more than one control rod withdrawn.

The revised pages in the Quad Cities Technical Specifications will delete the Unit 2 specific "a" pages, 2-1a and B2-3a, and replace them with pages applicable to both Units 1 and 2.

Revision to Thermal Limit Descriptions (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

This revision entails removing "for each type of fuel as a function of bundle/planar average exposure" from the Section 3 description of the APLHGR in the Quad Cities, Dresden, and LaSalle Technical Specifications. Thus the new description for all three site's Technical Specifications would state, "All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGR) shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT." The definition of Average Planar Exposure is also deleted from the Definitions section of the Technical Specifications for all three BWRs.

This revision entails removing "for each type of fuel as a function of AVERAGE PLANAR EXPOSURE" from the Section 3.11.D description of the LHGR in the Dresden Technical Specifications. Thus the new description for all three site's Technical Specifications would state, "The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed the STEADY STATE LINEAR HEAT GENERATION RATE (SLHGR) limits specified in the CORE OPERATING LIMITS REPORT." The definition of Average Planar Exposure is also deleted from the Definitions section of the Dresden Technical Specifications.

7. BASIS FOR THE REVISED REQUIREMENTS

Addition of SPC Revised Jet Pump Methodology (LaSalle Units 1 and 2)

The revised LOCA ECCS methodology (Reference 1), which has been submitted to the NRC and is currently under review, is a supplement to the NRC approved EXEM/BWR LOCA ECCS evaluation model (Reference 8). NRC approval has not been received for Reference 1, but is required prior to implementation of this proposed amendment. Because the revised jet pump methodology of Reference 1 will be used to determine APLHGR limits and because Generic Letter 88-16 indicates that Section 6.6.A.6.b of LaSalle's Technical Specifications is to include the "analytical methods used to determine the operating limits", it is appropriate that this reference be included in Section 6.6.A.6.b. Bases Section 3/4.2 also references the documents that describe the calculational models used to perform the LOCA analysis. Therefore, it is appropriate that this reference also be included in Bases Section 3/4.2.

Addition of SPC Generic Methodology for Application of ANFB Critical Power Correlation to Non-SPC Fuel (Quad Cities Units 1 and 2 and LaSalle Units 1 and 2)

In addition to SPC fuel, Quad Cities Units 1&2 and LaSalle Units 1&2 will be operating with previously exposed GE fuel. Because the generic methodology for applying the ANFB critical power correlation to coresident non-SPC fuel (Reference 3) will be used in establishing and monitoring MCPR limits for the coresident GE fuel, it is appropriate that this reference be included in Section 6.9.A.6.b of the Quad Cities Technical Specifications and Bases Section 2.1.2 and Section 6.6.A.6.b of the LaSalle Technical Specifications. This SPC topical report describes the methodology used to determine the additive constants and the associated uncertainty for application of the ANFB Critical Power Correlation to GE fuel. The additive constant uncertainty for the GE fuel is a parameter used in the calculation of a particular cycle's MCPR Safety Limit. Because the Safety Limit is used to determine the operating limit and because Generic Letter 88-16 indicates that Section 6.9.A.6.b of Quad Cities' Technical Specifications or 6.6.A.6.b of LaSalle's Technical Specifications is to include the "analytical methods used to determine the operating limits", it is appropriate that this reference be included. Because Bases Section 2.1.2 of LaSalle's Technical Specifications references the documents containing "The bases for the fuel-related uncertainties", it is also appropriate to add Reference 3 to LaSalle Bases Section 2.1.2.

Reference 3, which has been approved by the NRC, supersedes prior cycle specific references for Quad Cities Unit 2 Cycle 15 and LaSalle Unit 2 Cycle 8. This Technical Specification amendment proposes to remove these cycle specific references from Section 6.9.A.6.b of the Quad Cities Technical Specification and Bases of Section 2.1.2 of the LaSalle Unit 2 Technical Specifications and add Reference 3 to Section 6 of both Quad Cities and LaSalle Units 1 and 2 Technical Specifications.

Addition of SPC Topical for Revised ANFB Correlation Uncertainty (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

The ANFB Critical Power Correlation (Reference 6) is currently used in establishing the MCPR limits for Quad Cities Unit 2, Dresden Unit 2 and 3, and LaSalle Unit 2. It will be used for Quad Cities Unit 1 and LaSalle Unit 1 cores when SPC ATRIUM-9B fuel is loaded in upcoming cycles. Since the revised ATRIUM-9B ANFB additive constant uncertainty calculation is detailed in Reference 7, which has been submitted to the NRC and is currently under review, it is appropriate that Reference 7 be included in the list of methods in Section 6.9.A.6.b of the Dresden and Quad Cities Technical Specifications and Bases Section 2.1.2 and Section 6.6.A.6.b of the LaSalle Technical Specifications.

The new ATRIUM-9B additive constant uncertainty that will be used as a result of adding Reference 7 to Section 6 of Quad Cities, Dresden, and LaSalle's Technical Specifications is 0.0195. The basis of this additive constant uncertainty is a result of SPC increasing its ATRIUM-9B critical power test data base from 125 data points to 527 data points to cover a much wider range of pressures, mass fluxes, and axial power shapes. A statistical analysis was performed using these 527 points and documented in Reference 7.

Reference 7 was submitted to the NRC for review on April 18, 1997. Prior to being added to Dresden, Quad Cities, and LaSalle's Technical Specifications, Reference 7 must be NRC approved. Because the Safety Limit is used to determine the operating limit and because Generic Letter 88-16 indicates that Section 6.9.A.6.b of Quad Cities and Dresden's Technical Specifications and 6.6.A.6.b of LaSalle's Technical Specifications is to include the "analytical methods used to determine the operating limits", it is appropriate that this reference be included. It is also appropriate that Reference 7 be added to Bases Section 2.1.2 of LaSalle Technical Specifications, because Bases Section 2.1.2 of LaSalle's Technical Specifications references the documents containing "The bases for the fuel-related uncertainties".

Change to Minimum Critical Power Ratio Safety Limit (Quad Cities Units 1 and 2 and Dresden Units 2 and 3)

The basis for revising the MCPR Safety Limit for Quad Cities Unit 2 is the new Reference 7 methodology, which documents the calculation of a 0.0195 ATRIUM-9B additive constant uncertainty for the ATRIUM-9B critical power expanded data base. The current 0.029 ATRIUM-9B additive constant uncertainty is based on a conservative interim approach. This approach is to be used until the NRC approval of Reference 7 which is currently in review.

The basis for revising the MCPR Safety Limit for Quad Cities Unit 1 is due to the transition to SPC ATRIUM-9B fuel. Quad Cities Unit 1 Cycle 16 will be the first Unit 1 cycle with SPC ATRIUM-9B fuel. Therefore, Quad Cities Unit 1 Cycle 16 will not use the GEXL correlation to perform its MCPR Safety Limit calculations; however, References 3, 6, and 7 will be used. The MCPR Safety Limit that is supportable for Unit 1 is anticipated to be 1.09 or less. Cycle specific MCPR Safety Limit calculations are performed each reload to verify compliance with the MCPR Safety Limit in the Technical Specifications.

The basis for the removal of the paragraph in Bases Section 2.1.B of the Quad Cities Technical Specifications is that both the Unit 1 and Unit 2 MCPR Safety Limits are calculated using SPC methodology and are proposed to be changed to the same value, 1.09.

The basis for revising the MCPR Safety Limit for Dresden Units 2 and 3 is the expectation that future cycles may require a higher MCPR Safety Limit than 1.08 to support operation with ATRIUM-9B fuel. Increasing the MCPR Safety Limit to 1.09 provides some margin that will minimize the potential for a future Technical Specification amendment request.

Removal of Footnotes Limiting Operation with ATRIUM-9B Fuel Reloads (Quad Cities Unit 2 and Dresden Unit 3)

The removal of the footnotes in the Quad Cities Technical Specifications is justified by the NRC approval of Reference 3, which implements the generic application of the ANFB correlation to the coresident GE fuel. The removal of the footnotes in the Dresden Technical Specifications is also justified upon NRC approval of Reference 7 which describes the SPC calculation of the ATRIUM-9B additive constant uncertainty used in MCPR Safety Limit calculations. Thus removal of these footnotes is justified by the NRC approval of both the Reference 3 and Reference 7 methodologies.

The removal of the Unit 2 specific "a" pages, 2-1a and B2-3a, in the Quad Cities Technical Specifications is justified by the removal of the footnotes.

Revision to Thermal Limit Descriptions (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

The revised requirements provide flexibility for the exposure basis of the APLHGR limit. This revision allows the sites to utilize the appropriate exposure based APLHGR limits (which may be either bundle or average planar exposure based) and to specify them in the COLR. The revision also establishes consistent wording among the sites by utilizing the same definition for Quad Cities, Dresden, and LaSalle. Thus the revision provides flexibility for the APLHGR calculation, meets the guidelines of 10 CFR 50.46, and is consistent with NUREG 1433/1434.

The revised requirements also provide flexibility for the exposure basis of the LHGR limit at Dresden. This revision allows Dresden to specify the appropriate exposure basis for the LHGR limit in the COLR. This revision establishes consistent wording among the sites by utilizing a similar definition for Quad Cities, Dresden, and LaSalle. Thus the revision provides flexibility for the LHGR calculation and is consistent with NUREG 1433/1434.

8. SCHEDULE

ComEd requests approval of this Technical Specification amendment by February 15, 1998. After NRC approval, this Technical Specification amendment will be implemented at different times at the ComEd BWRs, based upon need. The implementation time for this Technical Specification amendment varies for each of the stations due to variances in outage schedules and operational status.

For Dresden, this amendment will be implemented concurrently with the Dresden Unit 2 Cycle 16 outage, which will be the first reload of SPC ATRIUM-9B for Unit 2 and will be the second reload of SPC ATRIUM-9B at Dresden Station. Dresden Unit 2 Cycle 16 is currently scheduled to begin operation on April 16, 1998. All changes proposed in this amendment will be included in the Dresden Technical Specifications prior to startup of Unit 2 Cycle 16.

For Quad Cities, this amendment will be implemented concurrently with the Quad Cities Unit 1 Cycle 16 outage, which will be the first reload of SPC ATRIUM-9B for Unit 1 and will be the second reload of SPC ATRIUM-9B at Quad Cities Station. Quad Cities Unit 1 Cycle 16 is currently scheduled to begin operation on October 15, 1998. All changes proposed in this amendment will be included in the Quad Cities Technical Specifications prior to startup of Unit 1 Cycle 16.

Because LaSalle Unit 2 Cycle 8, which contains a reload of SPC ATRIUM-9B fuel, will be the first cycle at LaSalle to need the Reference 1 revised jet pump methodology, this amendment is proposed to be implemented in LaSalle Unit 2 Technical Specifications prior to startup of Unit 2 Cycle 8. This will allow new APLHGR limits to be calculated that will support LaSalle Unit 2 Cycle 8 operation at full power. LaSalle Unit 1 Cycle 9 will be the first cycle for Unit 1 to contain a reload of SPC ATRIUM-9B fuel. Changes in this amendment will be included in the LaSalle Unit 1 Technical Specifications prior to startup of LaSalle Unit 1 Cycle 9. Currently, LaSalle Unit 1 Cycle 9 is projected to begin operation on December 17, 1998.

The requested approval date, February 15, 1998, was selected to support these scheduled cycle startup dates.

ATTACHMENT B

SUMMARY OF PROPOSED CHANGES TO THE

Quad Cities Units 1 and 2

Dresden Units 2 and 3

LaSalle Unit 1

LaSalle Unit 2

TECHNICAL SPECIFICATIONS

SUMMARY OF PROPOSED CHANGES

B. SUMMARY OF PROPOSED CHANGES

Quad Cities Units 1 and 2

Topic	*Affected Pages*	*Description of Change*
Delete Definition of Average Planar Exposure	I	The Table of Contents item for the definition of Average Planar Exposure is deleted.
Delete Definition of Average Planar Exposure	1-1	The definition of Average Planar Exposure is deleted.
Change to Minimum Critical Power Ratio Safety Limit	2-1	The MCPR Safety Limit for Unit 1 is increased to 1.09 (from 1.07) and the MCPR Safety Limit for Unit 2 is decreased to 1.09 (from 1.10).
Removal of Footnote for Cycle Specific MCPR Safety Limit	2-1a	The footnote stating that the Quad Cities Unit 2 MCPR Safety Limit is only applicable to Quad Cities Unit 2 Cycle 15 is deleted.
Removal of "a" pages	2-1a	Page 2-1a is deleted.
Removal of Footnote for Cycle Specific MCPR Safety Limit	B2-3a	The footnote stating that the Quad Cities Unit 2 MCPR Safety Limit is only applicable to Quad Cities Unit 2 Cycle 15 is deleted.
Deletion of Unit Specific MCPR Safety Limit Discussion	B2-3a	The paragraph that specified the MCPR Safety Limit for each Unit and what vendor's methodology is applicable to each unit is deleted.
Removal of "a" pages	B2-3a	Page B2-3a is deleted.
Removal of APLHGR Exposure Basis Requirements	3/4.11-1	The description of the APLHGR LCO is changed to not specify that APLHGR should be a function of average planar exposure.
Removal of Footnote for Cycle Specific MCPR Safety Limit Methodology	6-16a	The footnote stating that the MCPR methodology used to calculate the Q2C15 MCPR is only applicable to Unit 2 Cycle 15 is deleted.
Removal of Cycle Specific MCPR Methodology	6-16a	The Q2C15 cycle specific MCPR methodology used to calculate the Q2C15 MCPR is deleted.
Addition of SPC Topical for Application of the ANFB Correlation to Coresident Non-SPC Fuel	6-16a	The SPC MCPR methodology topical report for application of the ANFB correlation to coresident non-SPC fuel, EMF-1125(P)(A), Supplement 1, Appendix C, is added to the list of references.
Addition of SPC Topical Documenting the Additive Constant Uncertainties for the ATRIUM-9B Fuel	6-16a	The SPC topical report for the ATRIUM-9B additive constant uncertainties, ANFB-1125, Supplement 1, Appendix D, is added to the list of references.

ATTACHMENT B

SUMMARY OF PROPOSED CHANGES

Dresden Units 2 and 3

Topic	*Affected Pages*	*Description of Change*
Delete Definition of Average Planar Exposure	I	The Table of Contents item for the definition of Average Planar Exposure is deleted.
Delete Definition of Average Planar Exposure	1-1	The definition of Average Planar Exposure is deleted.
Change to Minimum Critical Power Ratio Safety Limit	2-1	The MCPR Safety Limit for Unit 2 and Unit 3 is increased to 1.09 from 1.08.
Removal of APLHGR Exposure Basis Requirements	3/4.11-1	The description of the APLHGR LCO is changed to not specify that APLHGR should be a function of bundle average exposure.
Removal of LHGR Exposure Basis Requirements	3/4.11-4	The description of the SLHGR LCO is changed to not specify that LHGR should be a function of average planar exposure.
Removal of Footnotes Limiting the use of ATRIUM-9B Reloads	5-5	Three footnotes that limit the use and design bases of ATRIUM-9B reloads in Unit 2 to Modes 3, 4, and 5, with no more than one control rod withdrawn are deleted.
Addition of SPC Topical documenting the Additive Constant Uncertainties for the ATRIUM-9B Fuel	6-15	The SPC topical report for the ATRIUM-9B additive constant uncertainties, ANF-1125, Supplement 1, Appendix D, is added to the list of references.

ATTACHMENT B

SUMMARY OF PROPOSED CHANGES

LaSalle Unit 1

Topic	*Affected Pages*	*Description of Change*
Delete Definition of Average Planar Exposure	I	The Index item for the definition of Average Planar Exposure is deleted.
Delete Definition of Average Planar Exposure	1-1	The definition of Average Planar Exposure is deleted.
Addition of SPC Topical Documenting the Additive Constant Uncertainties for the ATRIUM-9B Fuel	B2-2	The SPC Topical report for the ATRIUM-9B additive constant uncertainties, ANFB-1125, Supplement 1, Appendix D, is added to the list of references.
Addition of SPC Topical for Application of the ANFB Correlation to Coresident Non-SPC Fuel	B2-2	The SPC MCPR methodology topical report for application of the ANFB correlation to coresident non-SPC fuel, EMF-1125(P)(A) Supplement 1, Appendix C, is added to the list of references.
Removal of APLHGR Exposure Basis Requirements	3/4 2-1	The description of the APLHGR LCO is changed to not specify that APLHGR should be a function of average planar exposure.
Addition of SPC Jet Pump Model Revision LOCA Methodology	B3/4 2-5	The SPC topical report describing the revision to the jet pump model in the LOCA methodology, ANF-91-048(P), Supplement 1, is included in Reference 1 (ANF-91-048(P)(A)).
Addition of SPC Jet Pump Model Revision LOCA Methodology	6-25b	The SPC Topical report describing the revision to the jet pump model in the LOCA methodology, ANF-91-048(P), Supplement 1, is added to the list of references.
Addition of SPC Topical Documenting the Additive Constant Uncertainties for the ATRIUM-9B Fuel	6-25b	The SPC Topical report for the ATRIUM-9B additive constant uncertainties, ANFB-1125, Supplement 1, Appendix D, is added to the list of references.
Addition of SPC Topical for Application of the ANFB Correlation to Coresident Non-SPC Fuel	6-25b	The SPC MCPR methodology topical report for application of the ANFB correlation to coresident non-SPC fuel, EMF-1125(P)(A) Supplement 1, Appendix C, is added to the list of references.

ATTACHMENT B

SUMMARY OF PROPOSED CHANGES

LaSalle Unit 2

Topic	*Affected Pages*	*Description of Change*
Delete Definition of Average Planar Exposure	I	The Index item for the definition of Average Planar Exposure is deleted.
Delete Definition of Average Planar Exposure	1-1	The definition of Average Planar Exposure is deleted.
Removal of Cycle Specific MCPR Methodology	B 2-2	The L2C8 cycle specific MCPR methodology used to calculate the L2C8 MCPR Safety Limit is deleted.
Addition of SPC Topical Documenting the Additive Constant Uncertainties for the ATRIUM-9B Fuel	B2-2	The SPC Topical report for the ATRIUM-9B additive constant uncertainties, ANFB-1125, Supplement 1, Appendix D, is added to the list of references.
Addition of SPC Topical for Application of the ANFB Correlation to Coresident Non-SPC Fuel	B2-2	The SPC MCPR methodology topical report for application of the ANFB correlation to coresident non-SPC fuel, EMF-1125(P)(A) Supplement 1, Appendix C, is added to the list of references.
Removal of APLHGR Exposure Basis Requirements	3/4 2-1	The description of the APLHGR LCO is changed to not specify that APLHGR should be a function of average planar exposure.
Addition of SPC Jet Pump Model Revision LOCA Methodology	B3/4 2-5	The SPC topical report describing the revision to the jet pump model in the LOCA methodology, ANF-91-048(P), Supplement 1, is included in Reference 1 (ANF-91-048(P)(A)).
Addition of SPC Jet Pump Model Revision LOCA Methodology	6-25a	The SPC topical report describing the revision to the jet pump model in the LOCA methodology, ANF-91-048(P), Supplement 1, is added to the list of references.
Addition of SPC Topical Documenting the Additive Constant Uncertainties for the ATRIUM-9B Fuel	6-25a	The SPC topical report for the ATRIUM-9B additive constant uncertainties, ANFB-1125, Supplement 1, Appendix D, is added to the list of references.
Addition of SPC Topical for Application of the ANFB correlation to Coresident Non-SPC Fuel	6-25a	The SPC MCPR methodology topical report for application of the ANFB correlation to coresident non-SPC fuel, EMF-1125(P)(A) Supplement 1, Appendix C, is added to the list of references.

Attachment C

Marked Up Pages and Inserts for Quad Cities Technical Specifications

DEFINITIONS

SECTION PAGE

Section 1 DEFINITIONS

9709090106 970829
PDR ADOCK 05000237
P PDR

1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

Delete

AVERAGE PLANAR EXPOSURE (APE)

The AVERAGE PLANAR EXPOSURE (APE) shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATE(s) for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL

A CHANNEL shall be an arrangement of a sensor and associated components used to evaluate plant variables and generate a single protective action signal. A CHANNEL terminates and loses its identity where single action signals are combined in a TRIP SYSTEM or logic system.

CHANNEL CALIBRATION

A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the CHANNEL output such that it responds with the necessary range and accuracy to known values of the parameter which the CHANNEL monitors. The CHANNEL CALIBRATION shall encompass the entire CHANNEL including the required sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total CHANNEL steps such that the entire CHANNEL is calibrated.

CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment of CHANNEL behavior during operation by observation. This determination shall include, where possible, comparison of the CHANNEL indication and/or status with other indications and/or status derived from independent instrument CHANNEL(s) measuring the same parameter.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.A THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.

THERMAL POWER, High Pressure and High Flow

1.09

2.1.B The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.07 with the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow. During single recirculation loop operation, this MCPR limit shall be increased by 0.01.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

With MCPR less than the above applicable limit and the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.

2.1 SAFETY LIMITS

REMOVE THIS PAGE

THERMAL POWER, Low Pressure or Low Flow

2.1.A THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.

THERMAL POWER, High Pressure and High Flow

2.1.B The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.07 for Unit 1 and 1.10* for Unit 2 with the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow. During single recirculation loop operation, this MCPR limit shall be increased by 0.01.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

With MCPR less than the above applicable limit and the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.

* Applicable to Unit 2 for cycle 15 only.

DELETE

 REMOVE THIS PAGE

approach. Much of the data indicates that BWR fuel can survive for an extended period in an environment of transition boiling.

The Unit 1 MCPR Safety Limit is 1.07, based on General Electric methods for calculating the MCPR Safety Limit. The Unit 2 MCPR Safety Limit is 1.10*, based on Siemens Power Corporation (SPC) methods for calculating the MCPR Safety Limit.

DELETE

2.1.C Reactor Coolant System Pressure

The Safety Limit for the reactor coolant system pressure has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor coolant system integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The reactor coolant system pressure Safety Limit of 1345 psig, as measured by the vessel steam space pressure indicator, is equivalent to 1375 psig at the lowest elevation of the reactor vessel. The 1375 psig value is derived from the design pressures of the reactor pressure vessel and coolant system piping. The respective design pressures are 1250 psig at 575°F and 1175 psig at 560°F. The pressure Safety Limit was chosen as the lower of the pressure transients permitted by the applicable design codes, ASME Boiler and Pressure Vessel Code Section III for the pressure vessel, and USASI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% x 1250 = 1375 psig), and the USASI Code permits pressure transients up to 20% over design pressure (120% x 1175 = 1410 psig). The Safety Limit pressure of 1375 psig is referenced to the lowest elevation of the reactor vessel. The design pressure for the recirculation suction line piping (1175 psig) was chosen relative to the reactor vessel design pressure. Demonstrating compliance of peak vessel pressure with the ASME overpressure protection limit (1375 psig) assures compliance of the suction piping with the USASI limit (1410 psig). Evaluation methodology to assure that this Safety Limit pressure is not exceeded for any reload is documented by the specific fuel vendor. The design basis for the reactor pressure vessel makes evident the substantial margin of protection against failure at the safety pressure limit of 1375 psig. The vessel has been designed for a general membrane stress no greater than 26,700 psi at an internal pressure of 1250 psig; this is a factor of 1.5 below the yield strength of 40,100 psi at 575°F. At the pressure limit of 1375 psig, the general membrane stress will only be 29,400 psi, still safely below the yield strength.

The relationships of stress levels to yield strength are comparable for the primary system piping and provides similar margin of protection at the established pressure Safety Limit.

The normal operating pressure of the reactor coolant system is nominally 1000 psig. Both pressure relief and safety relief valves have been installed to keep the reactor vessel peak pressure below 1375 psig. However no credit is taken for relief valves during the postulated full closure of all MSIVs without a direct (valve position switch) scram. Credit, however, is taken for the neutron flux scram. The indirect flux scram and safety valve actuation provide adequate margin below the allowable peak vessel pressure of 1375 psig.

* Applicable to Unit 2 cycle 15 only. DELETE

3.11 - LIMITING CONDITIONS FOR OPERATION

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

delete

All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGR) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY:

OPERATIONAL MODE 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT:

1. Initiate corrective ACTION within 15 minutes, and

2. Restore APLHGR to within the required limit within 2 hours.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

A. AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The APLHGRs shall be verified to be equal to or less than the limits specified in the CORE OPERATING LIMITS REPORT.

1. At least once per 24 hours,

2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

3. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

4. The provisions of Specification 4.0.D are not applicable.

(14) ANFB Critical Power Correlation, ANF-1125(P)(A) and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, April 1990.

(15) Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, ANF-524(P)(A), Revision 2, Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation, November 1990.

(16) COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analyses, ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, Advanced Nuclear Fuels Corporation, August 1990.

(17) Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model, ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation, January 1993.

(18) Commonwealth Edison Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods," Revision 0, Supplements 1 and 2, December 1991, March 1992, and May 1992, respectively; SER letter dated March 22, 1993.

Delete

(19)*ComEd letter, "ComEd Response to NRC Staff Request for Additional Information (RAI) Regarding the Application of Siemens Power Corporation ANFB Critical Power Correlation to Coresident General Electric Fuel for LaSalle Unit 2 Cycle 8 and Quad Cities Unit 2 Cycle 15, NRC Docket No.'s 50-373/374 and 50-254/265", J.B. Hosmer to U.S. NRC, July 2, 1996, transmitting the topical report, Application of the ANFB Critical Power Correlation to Coresident GE Fuel for Quad Cities Unit 2 Cycle 15, EMF-96-051(P), Siemens Power Corporation - Nuclear Division, May 1996, and related information.

Insert A →

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.

Delete

*Applicable to Unit 2 for cycle 15 only.

INSERT A

QUAD CITIES Section 6.9.A.6.b Technical Specifications Insert

(19) ANFB Critical Power Correlation Application for Coresident Fuel, EMF-1125(P)(A), Supplement 1, Appendix C, Siemens Power Corporation, August 1997.

(20) ANFB Critical Power Correlation Uncertainty for Limited Data Sets, ANF-1125(P)(A), Supplement 1, Appendix D, Siemens Power Corporation, (DATE TO BE DETERMINED).

Attachment D

Marked Up Pages and Inserts for Dresden Technical Specifications

DEFINITIONS

SECTION PAGE

1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

Delete

AVERAGE PLANAR EXPOSURE (APE)

The AVERAGE PLANAR EXPOSURE (APE) shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATE(s) for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL

A CHANNEL shall be an arrangement of a sensor and associated components used to evaluate plant variables and generate a single protective action signal. A CHANNEL terminates and loses its identity where single action signals are combined in a TRIP SYSTEM or logic system.

CHANNEL CALIBRATION

A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the CHANNEL output such that it responds with the necessary range and accuracy to known values of the parameter which the CHANNEL monitors. The CHANNEL CALIBRATION shall encompass the entire CHANNEL including the required sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total CHANNEL steps such that the entire CHANNEL is calibrated.

CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment of CHANNEL behavior during operation by observation. This determination shall include, where possible, comparison of the CHANNEL indication and/or status with other indications and/or status derived from independent instrument CHANNEL(s) measuring the same parameter.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.A THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.

THERMAL POWER, High Pressure and High Flow

1.09

2.1.B The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.08 with the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow. During single recirculation loop operation, this MCPR limit shall be increased by 0.01.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

With MCPR less than the above applicable limit and the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.

3.11 - LIMITING CONDITIONS FOR OPERATION

A. AVERAGE PLANAR LINEAR HEAT GENERATION RATE

All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGR) for each type of fuel as a function of bundle average exposure shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

Delete

APPLICABILITY:

OPERATIONAL MODE 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT:

1. Initiate corrective action within 15 minutes, and
2. Restore APLHGR to within the required limit within 2 hours.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

A. AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The APLHGRs shall be verified to be equal to or less than the limits specified in the CORE OPERATING LIMITS REPORT.

1. At least once per 24 hours,
2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
3. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.
4. The provisions of Specification 4.0.D are not applicable.

3.11 - LIMITING CONDITIONS FOR OPERATION

D. STEADY STATE LINEAR HEAT GENERATION RATE

DELETE

The LINEAR HEAT GENERATION RATE (LHGR) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the STEADY STATE LINEAR HEAT GENERATION RATE (SLHGR) limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY:

OPERATIONAL MODE 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an LHGR exceeding the SLHGR limits specified in the CORE OPERATING LIMITS REPORT:

1. Initiate corrective ACTION within 15 minutes, and
2. Restore the LHGR to within the SLHGR limit within 2 hours.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

D. STEADY STATE LINEAR HEAT GENERATION RATE

The SLHGR shall be determined to be equal to or less than the limit:

1. At least once per 24 hours,
2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
3. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for SLHGR.
4. The provisions of Specification 4.0.D are not applicable.

5.0 DESIGN FEATURES

5.3 REACTOR CORE

Fuel Assemblies

Delete

5.3.A The reactor core shall contain 724 fuel assemblies[1,2]. Each assembly consists of a matrix of Zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. The assemblies may contain water rods or a water box. Limited substitutions of Zircaloy or ZIRLO or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases[3]. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

Delete

Control Rod Assemblies

5.3.B The reactor core shall contain 177 cruciform shaped control rod assemblies. The control material shall be boron carbide powder (B_4C) and/or hafnium metal. The control rod assembly shall have a nominal axial absorber length of 143 inches.

Delete

1 ATRIUM-9B fuel with exception of lead test assemblies is only allowed in the reactor core in Operational Modes 3, 4 and 5, and with no more than one control rod withdrawn, for Unit 2 only.

2 Operation in all modes with ATRIUM-9B fuel is allowed for Dresden, Unit 3, Cycle 15, only.

3 The design bases applicable to ATRIUM-9B fuel are those which are applicable to Operational Modes 3, 4, and 5, for Unit 2 only.

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

(7) XN-NF-82-06(P)(A), Qualification of Exxon Nuclear Fuel for Extended Burnup Supplement 1 Extended Burnup Qualification of ENC 9x9 BWR Fuel, Supplement 1, Revision 2, Advanced Nuclear Fuels Corporation, May 1988.

(8) ANF-89-14(P)(A), Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advance Nuclear Fuels Corporation 9x9-IX and 9x9-9X BWR Reload Fuel, Revision 1 and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, October 1991.

(9) ANF-89-98(P)(A), Generic Mechanical Design Criteria for BWR Fuel Designs, Revision 1 and Revision 1 Supplement 1, Advanced Nuclear Fuels Corporation, May 1995.

(10) ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model, Advanced Nuclear Fuels Corporation, January 1993.

(11) 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).

Insert B

INSERT B

DRESDEN Section 6.9.A.6.b Technical Specification Insert

(12) ANF-1125(P)(A), ANFB Critical Power Correlation Uncertainty For Limited Data Sets, Supplement 1, Appendix D, Siemens Power Corporation, (DATE TO BE DETERMINED).

Attachment E

Marked Up Pages and Inserts for LaSalle Unit 1 Technical Specifications

DELETED

DEFINITIONS

SECTION

Delete

1.0 DEFINITIONS

DELETED The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

AVERAGE PLANAR EXPOSURE DELETE

1.2 The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle. Delete

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.3 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is tested.

2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using the ANF Critical Power Methodology for boiling water reactors (Reference 1) which is a statistical model that combines all of the uncertainties in operation parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the SPC-developed ANFB critical power correlation.

7 Replace

The bases for the uncertainties in system-related parameters are presented in NEDO-20340, Reference 2. The bases for the fuel-related uncertainties are found in References 1, 3-5. The uncertainties used in the analyses are provided in the cycle-specific transient analysis parameters document.

1. Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, XN-NF-524(P)(A) Revision 2 and Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation, November 1990.

2. Process Computer Performance Evaluation Accuracy, NEDO-20340 and Amendment 1, General Electric Company, June 1974 and December 1974, respectively.

3. ANFB Critical Power Correlation, ANF-1125(P)(A), and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, April 1990.

4. Advanced Nuclear Fuels Methodology for Boiling Water Reactors, XN-NF-80-19(P)(A) Volume 1 Supplement 3, Supplement 3 Appendix F, and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.

5. Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, March 1983.

Insert C

INSERT C

LASALLE UNIT 1 Bases Section 2.1.2 Technical Specifications Insert

6. ANFB Critical Power Correlation Application for Coresident Fuel, EMF-1125(P)(A), Supplement 1, Appendix C, Siemens Power Corporation, August 1997.

7. ANFB Critical Power Correlation Uncertainty for Limited Data Sets, ANF-1125(P)(A), Supplement 1, Appendix D, Siemens Power Corporation, (DATE TO BE DETERMINED).

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

Delete

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) ~~for each type of fuel as a function of AVERAGE PLANAR EXPOSURE~~ shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits specified in the CORE OPERATING LIMITS REPORT.

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

3/4.2.4 LINEAR HEAT GENERATION RATE

GE Fuel

The specification assures that the LINEAR HEAT GENERATION RATE (LHGR) in any rod is less than the design linear heat generation even if fuel pellet densification is postulated. The effects of fuel densification are discussed in the General Electric Standard Application for Reactor Fuel (GESTAR), NEDE-24011-P-A. The GESTAR discusses the methods used to ensure LHGR remains below the design limit.

SPC Fuel

The Linear Heat Generation Rate (LHGR) is a measure of the heat generation rate per unit length of a fuel rod in a fuel assembly at any axial location. LHGR limits are specified to ensure that fuel integrity limits are not exceeded during normal operation or anticipated operational occurrences (AOOs). Operation above the LHGR limit followed by the occurrence of an AOO could potentially result in fuel damage and subsequent release of radioactive material. Sustained operation in excess of the LHGR limit could also result in exceeding the fuel design limits. The failure mechanism prevented by the LHGR limit that could cause fuel damage during AOOs is rupture of the fuel rod cladding caused by strain from the expansion of the fuel pellet. One percent plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur. Fuel design evaluations are performed to demonstrate that the mechanical design limits are not exceeded during continuous operation with LHGRs up to the limit defined in the CORE OPERATING LIMITS REPORT. The analysis also includes allowances for short term transient operation above the LHGR limit.

At reduced power and flow conditions, the LHGR limit may need to be reduced to ensure adherence to the fuel mechanical design bases during limiting transients. At reduced power and flow conditions, the LHGR limit is reduced (multiplied) using the smaller of either the flow-dependent LHGR factor ($LHGRFAC_f$) or the power-dependent LHGR factor ($LHGRFAC_p$) corresponding to the existing core flow and power. **The $LHGRFAC_f$ multipliers are used to protect the** core during slow flow runout transients. The $LHGRFAC_p$ multipliers are used to protect the core during plant transients other than core flow transients. The applicable $LHGRFAC_f$ ~~and $LHGRFAC_p$ multipliers are specified in the CORE~~ OPERATING ~~LIMITS~~ REPORT.

INSERT: and BWR Jet Pump Model Revision for RELAX, ANF-91-048(P)(A), Supplement 1, Siemens Power Corporation, DATE TO BE DETERMINED.

References:

1. Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR ECCS Evaluation Model, ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation, January 1993.

2. Exxon Nuclear Methodology for Boiling Water Reactors, Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A), Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, March 1983.

3. Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description, XN-NF-80-19(P)(A), Volume 3 Revision 2, Exxon Nuclear Company, January 1987.

Core Operating Limits Report (Continued)

(16) Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A), Revision 2 Supplements 1, 2, and 3, Exxon Nuclear Company, March 1986.

(17) Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A), Revision 1 and Revision 1 Supplement 1, Advanced Nuclear Fuels Corporation, May 1995.

(18) NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (latest approved revision).

(19) Commonwealth Edison Topical Report NFSR-0085, "Benchmark of BWR Nuclear Design Methods," (latest approved revision).

(20) Commonwealth Edison Topical Report NFSR-0085, Supplement 1, "Benchmark of BWR Nuclear Design Methods - Quad Cities Gamma Scan Comparisons," (latest approved revision).

(21) Commonwealth Edison Topical Report NFSR-0085, Supplement 2, "Benchmark of BWR Nuclear Design Methods - Neutronic Licensing Analyses," (latest approved revision).

(22) Commonwealth Edison Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods," Revision 0, Supplements 1 and 2, December 1991, March 1992, and May 1992, respectively; SER letter dated March 22, 1993.

Insert D

INSERT D

LASALLE UNIT 1 Section 6.6.A.6.b Technical Specifications Insert

(23) BWR Jet Pump Model Revision for RELAX, ANF-91-048(P)(A), Supplement 1, Siemens Power Corporation, (DATE TO BE DETERMINED).

(24) ANFB Critical Power Correlation Application for Coresident Fuel, EMF-1125(P)(A), Supplement 1, Appendix C, Siemens Power Corporation, August 1997.

(25) ANFB Critical Power Correlation Uncertainty for Limited Data Sets, ANF-1125(P)(A), Supplement 1, Appendix D, Siemens Power Corporation, (DATE TO BE DETERMINED).

Attachment F

Marked up Pages and Inserts for LaSalle Unit 2 Technical Specifications

DELETED

DEFINITIONS

SECTION

DELETED

1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

AVERAGE PLANAR EXPOSURE DELETE

Delete

1.2 The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.3 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is tested.

2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using the ANF Critical Power Methodology for boiling water reactors (Reference 1) which is a statistical model that combines all of the uncertainties in operation parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the SPC-developed ANFB critical power correlation.

7 Replace

The bases for the uncertainties in system-related parameters are presented in NEDO-20340, Reference 2. The bases for the fuel-related uncertainties are found in References 1, 3-~~6~~. The uncertainties used in the analyses are provided in the cycle-specific transient analysis parameters document.

1. Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, XN-NF-524 (P)(A) Revision 2, and Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation, November 1990.

2. Process Computer Performance Evaluation Accuracy, NEDO-20340 and Amendment 1, General Electric Company, June 1974 and December 1974, respectively.

3. ANFB Critical Power Correlation, ANF-1125 (P)(A), and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, April 1990.

4. Advanced Nuclear Fuels Methodology for Boiling Water Reactors, XN-NF-80-19 (P)(A) Volume 1 Supplement 3, Supplement 3 Appendix F, and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.

5. Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, March 1983.

6. "Application of the ANFB Critical Power Correlation to Coresident GE Fuel for LaSalle Unit 2 Cycle 8," EMF-96-021(P), Revision 1, Siemens Power Corporation, February 1996; NRC SER letter dated September 26, 1996.

Delete

Insert E

INSERT E

LASALLE UNIT 2 Bases Section 2.1.2 Technical Specifications Insert

6. ANFB Critical Power Correlation Application for Coresident Fuel, EMF-1125(P)(A), Supplement 1, Appendix C, Siemens Power Corporation, August 1997.

7. ANFB Critical Power Correlation Uncertainty for Limited Data Sets, ANF-1125(P)(A), Supplement 1, Appendix D, Siemens Power Corporation, (DATE TO BE DETERMINED).

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

Delete

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits specified in the CORE OPERATING LIMITS REPORT.

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

BASES

3/4.2.4 LINEAR HEAT GENERATION RATE (Continued)

fuel damage caused by overstraining of the fuel cladding is not expected to occur. Fuel design evaluations are performed to demonstrate that the mechanical design limits are not exceeded during continuous operation with LHGRs up to the limit defined in the CORE OPERATING LIMITS REPORT. The analysis also includes allowances for short term transient operation above the LHGR limit.

At reduced power and flow conditions, the LHGR limit may need to be reduced to ensure adherence to the fuel mechanical design bases during limiting transients. At reduced power and flow conditions, the LHGR limit is reduced (multiplied) using the smaller of either the flow dependent LHGR factor ($LHGRFAC_f$) or the power-dependent LHGR factor ($LHGRFAC_p$) corresponding to the existing core flow and power. The $LHGRFAC_f$ multipliers are used to protect the core during slow flow runout transients. The $LHGRFAC_p$ multipliers are used to protect the core during plant transients other than core flow transients. The applicable $LHGRFAC_f$ and $LHGRFAC_p$ multipliers are specified in the CORE OPERATING LIMITS REPORT.

References:

1. Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR ECCS Evaluation Model, ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation, January 1993.

2. Exxon Nuclear Methodology for Boiling Water Reactors, Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A), Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, March 1983.

3. Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description, XN-NF-80-19(P)(A), Volume 3 Revision 2, Exxon Nuclear Company, January 1987.

4. Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A) Revision 2 Supplements 1, 2, and 3, Exxon Nuclear Company, March 1986.

5. COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses, ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, Advanced Nuclear Fuels Corporation, August 1990.

INSERT

and BWR Jet Pump Model Revision for RELAX, ANF-91-048 (P)(A), Supplement 1, Siemens Power Corporation, (DATE TO BE DETERMINED).

Core Operating Limits Report (Continued)

(9) Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, XN-NF-85-67(P)(A) Revision 1, Exxon Nuclear Company, September 1986.

(10) Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels Corporation 9x9-IX and 9x9-9X BWR Reload Fuel, ANF-89-014(P)(A), Revision 1 and Supplements 1 and 2, October 1991.

(11) Volume 1 - STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain, Volume 2 - STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain, Code Qualification Report, EMF-CC-074(P)(A), Siemens Power Corporation, July 1994.

(12) RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model, XN-NF-81-58(P)(A), Revision 2 Supplements 1 and 2, Exxon Nuclear Company, March 1984.

(13) XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis, XN-NF-84-105(P)(A), Volume 1 and Volume 1 Supplements 1 and 2; Volume 1 Supplement 4, Advanced Nuclear Fuels Corporation, February 1987 and June 1988, respectively.

(14) Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model, ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation, January 1993.

(15) Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, Richland, WA 99352, March 1983.

(16) Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A), Revision 2 Supplements 1, 2, and 3, Exxon Nuclear Company, March 1986.

(17) Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A), Revision 1 and Revision 1 Supplement 1, Advanced Nuclear Fuels Corporation, May 1995.

(18) NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (latest approved revision).

(19) Commonwealth Edison Topical Report NFSR-0085, "Benchmark of BWR Nuclear Design Methods," (latest approved revision).

(20) Commonwealth Edison Topical Report NFSR-0085, Supplement 1, "Benchmark of BWR Nuclear Design Methods - Quad Cities Gamma Scan Comparisons," (latest approved revision).

(21) Commonwealth Edison Topical Report NFSR-0085, Supplement 2, "Benchmark of BWR Nuclear Design Methods - Neutronic Licensing Analyses," (latest approved revision).

(22) Commonwealth Edison Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods," Revision 0, Supplements 1 and 2, December 1991, March 1992, and May 1992, respectively; SER letter dated March 22, 1993.

Insert F

INSERT F

LASALLE UNIT 2 Section 6.6.A.6.b Technical Specifications Insert

(23) BWR Jet Pump Model Revision for RELAX, ANF-91-048(P)(A), Supplement 1, Siemens Power Corporation, (DATE TO BE DETERMINED).

(24) ANFB Critical Power Correlation Application for Coresident Fuel, EMF-1125(P)(A), Supplement 1, Appendix C, Siemens Power Corporation, August 1997.

(25) ANFB Critical Power Correlation Uncertainty for Limited Data Sets, ANF-1125(P)(A), Supplement 1, Appendix D, Siemens Power Corporation, (DATE TO BE DETERMINED).

G. EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

Adding References 1 and 7 to Technical Specification Section 6 and applying these methods at ComEd BWRs is evaluated for significant hazards consideration in this section. These documents have been submitted to the NRC under separate correspondence. References 1 and 7 are in NRC review, and require approval to be inserted into Section 6.

ComEd has evaluated the proposed Technical Specification amendment and determined it does not represent a significant hazards consideration. Based on the criteria for defining a significant hazard consideration established in 10CFR50.92(c), operation of Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2, in accordance with the proposed amendments, will not represent a significant hazards consideration for the following reasons:

These changes do not:

1. **Involve a significant increase in the probability or consequences of an accident previously evaluated.**

   The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. Limits have been established consistent with NRC approved methods to ensure that fuel performance during normal, transient, and accident conditions is acceptable. These changes do not affect the operability of plant systems, nor do they compromise any fuel performance limits.

Addition of SPC Revised Jet Pump Methodology (LaSalle Units 1 and 2)

The Reference 1 methodology to be added to the Technical Specifications is used as part of the LOCA analysis and does not introduce physical changes to the plant. The Reference 1 revised jet pump model changes the calculational behavior of the jet pump under reversed drive flow conditions. The revised jet pump model methodology makes the LOCA model behave more realistically and calculates small break LOCA PCTs that are comparable to the large break LOCA results. Therefore, this change only affects the methodology for analyzing the LOCA event and determining the protective APLHGR limits. The Technical Specification requirements for monitoring APLHGR are not affected by this change. The revised method will result in higher APLHGR limits, thus the SPC fuel will be allowed to operate at higher nodal powers. The approved methodology, however, still protects the fuel performance limits specified by 10CFR50.46. Therefore, the probability or consequences of an accident previously evaluated will not change.

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

Addition of SPC Generic Methodology for Application of ANFB Critical Power Correlation to Non-SPC Fuel (Quad Cities Units 1 and 2 and LaSalle Units 1 and 2)

The probability or consequences of a previously evaluated accident are not increased by adding Reference 3 to Section 6.9.A.6.b of the Quad Cities Technical Specifications and Bases Section 2.1.2 and Section 6.6.A.6.b of the LaSalle Technical Specifications. Reference 3 determines the additive constants and the associated uncertainty for application of the ANFB correlation to the coresident GE fuel. Therefore, it provides data that is used in the determination of the MCPR Safety Limit. This approved methodology for applying the ANFB critical power correlation to the GE fuel will protect the fuel from boiling transition. Operational MCPR limits will also be applied to ensure that the MCPR Safety Limit is protected during all modes of operation and anticipated operational occurrences. Because Reference 3 contains conservative methods and calculations and because the operability of plant systems designed to mitigate any consequences of accidents have not changed, the probability or consequences of an accident previously evaluated will not increase.

Addition of SPC Topical for Revised ANFB Correlation Uncertainty (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

The probability or consequences of a previously evaluated accident is not increased by adding Reference 7 to Section 6.9.A.6.b of the Quad Cities and Dresden Technical Specifications and Bases Section 2.1.2 and Section 6.6.A.6.b of the LaSalle Technical Specifications. Reference 7 documents the additive constant uncertainty for SPC ATRIUM-9B fuel design with an internal water channel. This methodology is used to determine an input to the MCPR Safety Limit calculations, which ensures that more than 99.9% of the fuel rods avoid transition boiling during normal operation as well as anticipated operational occurrences. This change does not require any physical plant modifications, physically affect any plant components, or entail changes in plant operation. This methodology for determining the ATRIUM-9B additive constant uncertainty for the MCPR Safety Limit calculation will continue to support protecting the fuel from boiling transition. Operational MCPR limits will be applied to ensure the MCPR Safety Limit is not violated during all modes of operation and anticipated operational occurrences. Therefore, no individual precursors of an accident are affected and the operability of plant systems designed to mitigate the probability of consequences of an accident previously evaluated are not affected by these changes.

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

Change to Minimum Critical Power Ratio Safety Limit (Quad Cities Units 1 and 2 and Dresden Units 2 and 3)

Changing the MCPR Safety Limit at Quad Cities Units 1 and 2 and Dresden Units 2 and 3 will not increase the probability of an accident previously evaluated. This change implements the MCPR Safety Limits resulting from the SPC ANFB critical power correlation methodology using a revised additive constant uncertainty from Reference 7. The MCPR Safety Limit of 1.09 that is proposed for Quad Cities Units 1 and 2 and Dresden Units 2 and 3 is anticipated to be conservative and acceptable for future cycles. Cycle specific MCPR Safety Limit calculations will be performed, consistent with SPC's approved methodology, to confirm the appropriateness of the MCPR Safety Limit. Additionally, operational MCPR limits will be applied that will ensure the MCPR Safety Limit is not violated during all modes of operation and anticipated operational occurrences. Changing the MCPR Safety Limit will not alter any physical systems or operating procedures. The MCPR Safety Limit is set to 1.09, which is the CPR value where less than 0.1% of the rods in the core are expected to experience boiling transition. This safety limit is expected to be applicable for future cycles of ATRIUM-9B at Dresden and Quad Cities. Therefore the probability or consequences of an accident will not increase.

Removal of Footnotes Limiting Operation with ATRIUM-9B Fuel Reloads (Quad Cities Unit 2 and Dresden Unit 3)

The removal of footnotes from the Quad Cities and Dresden Technical Specifications does not involve any significant increase in the probability or consequences of an accident previously evaluated. The footnotes were added to clarify that cycle specific methods were used until the generic methodology was approved by the NRC. Since the NRC has approved SPC's generic methodology for application of the ANFB correlation to the coresident GE fuel (Reference 3) and SPC has addressed the concerns regarding the database used to calculate the ATRIUM-9B additive constant uncertainties (Reference 7), the footnotes are no longer necessary. The removal of the Unit 2 specific "a" pages, 2-1a and B2-3a, in the Quad Cities Technical Specifications is justified by the removal of the footnotes. Therefore, removing these footnotes and "a" pages does not require any physical plant modifications, nor does it physically affect any plant components or entail changes in plant operation. Therefore, the probability or consequences of an accident previously evaluated is not expected to increase.

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

Revision to Thermal Limit Descriptions (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

The revision to the Section 3 Technical Specification description of the APLHGR limits has no implications on accident analysis or plant operations. The purpose of the revision is to allow flexibility for the MAPLHGR limits and their exposure basis to be specified in the COLR and to establish consistency with approved methodologies currently utilized by Siemens Power Corporation, which calculates MAPLHGR limits based on bundle or planar average exposures. This revision also provides for consistency in the APLHGR limit Technical Specification wording between the ComEd BWRs. The revision to the 3.11.D SLHGR Technical Specification for Dresden also has no implications on accident analysis or plant operations. The purpose of this revision is to allow flexibility for the LHGR limits and their exposure basis to be specified in the COLR. This revision makes the Dresden LHGR definition consistent with NUREG 1433/1434 wording. The definition of the Average Planar Exposure is deleted, because the exposure basis of the APLHGR is being removed. Therefore, no plant equipment or processes are affected by this change. Thus, there is no alteration in the probability or consequences of an accident previously evaluated.

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

Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications to the plant configuration, including changes in allowable modes of operation. This Technical Specification submittal does not involve any modifications to the plant configuration or allowable modes of operation. No new precursors of an accident are created and no new or different kinds of accidents are created. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Addition of SPC Revised Jet Pump Methodology (LaSalle Units 1 and 2)

The revised jet pump model methodology will be used to analyze the LOCA for LaSalle Units 1 and 2, and does not introduce any physical changes to the plant or the processes used to operate the plant. This change only affects the methods used to analyze the LOCA event and determine the MAPLHGR limits. Therefore, the possibility of a new or different kind of accident is not created.

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

Addition of SPC Generic Methodology for Application of ANFB Critical Power Correlation to Non-SPC Fuel (Quad Cities Units 1 and 2 and LaSalle Units 1 and 2)

Addition of the generic methodology for the application of the ANFB critical power correlation to GE fuel in Section 6.9.A.6.b of the Quad Cities Technical Specifications and Bases Section 2.1.2 and Section 6.6.A.6.b of the LaSalle Technical Specifications does not introduce any physical changes to the plant, the processes used to operate the plant, or allowable modes of operation. This change only involves adding an NRC approved methodology, which is used to determine the additive constants and additive constant uncertainty for GE fuel, to Section 6 of the Technical Specifications. Therefore, no new precursors of an accident are created and no new or different kinds of accidents are created.

Addition of SPC Topical for Revised ANFB Correlation Uncertainty (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

Addition of the Reference 7 methodology to Section 6.9.A.6.b of the Quad Cities and Dresden Technical Specifications and Bases Section 2.1.2 and Section 6.6.A.6.b of the LaSalle Technical Specifications will not create the possibility of a new or different kind of accident from any accident previously evaluated. This methodology describes the calculation of an input to the MCPR Safety Limit - the ATRIUM-9B additive constant uncertainty. Therefore, no new precursors of an accident are created and no new or different kinds of accidents are created.

Change to Minimum Critical Power Ratio Safety Limit (Quad Cities Units 1 and 2 and Dresden Units 2 and 3)

Changing the MCPR Safety Limit will not create the possibility of a new accident from an accident previously evaluated. This change will not alter or add any new equipment or change modes of operation. The MCPR Safety Limit is established to ensure that 99.9% of the rods avoid boiling transition.

The MCPR Safety Limit is changing for Quad Cities Unit 1 due to the transition to SPC ATRIUM-9B fuel and SPC methodologies. The MCPR Safety Limit is changing for Quad Cities Unit 2 due to the Reference 7 methodology, which documents a 0.0195 ATRIUM-9B additive constant uncertainty and supports a 1.09 MCPR Safety Limit. This MCPR Safety Limit is lower than the current MCPR Safety Limit for Quad Cities Unit 2, 1.10, which is based on a higher interim conservative additive constant uncertainty of 0.029. The lower ATRIUM-9B additive constant uncertainty results in the lower MCPR Safety Limit for Quad Cities Unit 2. The new MCPR Safety Limit for Dresden Units 2 and 3, 1.09, is greater than the current value at Dresden Units 2 and 3 and is being increased now in anticipation of bounding future reloads of ATRIUM-9B. Therefore, no new accidents are created that are different from any accident previously evaluated.

Removal of Footnotes Limiting Operation with ATRIUM-9B Fuel Reloads (Quad Cities Unit 2 and Dresden Unit 3)

The removal of the footnotes from the Quad Cities and Dresden Technical Specifications does not create a new or different kind of accident from any accident previously evaluated. The removal of the footnotes does not affect plant systems or operation. The footnotes were temporarily established to implement a conservative cycle specific MCPR Safety Limit until the SPC generic methodology was approved. With the approval of the generic Reference 3 methodology and the anticipated approval of the Reference 7 additive constant uncertainty methodology, these footnotes are no longer applicable. The removal of the Unit 2 specific "a" pages, 2-1a and B2-3a, in the Quad Cities Technical Specifications which is justified by the removal of the footnotes, also does not create a new or different kind of accident from any accident previously evaluated.

Revision to Thermal Limit Descriptions (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle 1 and 2)

The revision of the APLHGR and LHGR limit descriptions will not create the possibility of a new or different kind of accident from any accident previously evaluated. This revision will not alter any plant systems, equipment, or physical conditions of the site. **This revision allows the flexibility of the APLHGR and the LHGR limits to be specified in the COLR and to maintain consistency with the calculated results of methodologies** currently used to determine the APLHGR. The definition of the Average Planar Exposure is deleted, because it is being removed from LHGR and APLHGR Technical Specifications.

3. Involve a significant reduction in the margin of safety for the following reasons:

Addition of SPC Revised Jet Pump Methodology (LaSalle Units 1 and 2)

The revised jet pump model methodology, and the MAPLHGRs, resulting from the revised jet pump methodology, will continue to ensure fuel design criteria and 10CFR50.46 compliance. The results of LOCA analyses performed with this methodology must continue to comply with the requirements of 10CFR50.46. Therefore, there is no significant reduction in the margin of safety.

Addition of SPC Generic Methodology for Application of ANFB Critical Power Correlation to Non-SPC Fuel (Quad Cities Units 1 and 2 and LaSalle Units 1 and 2)

The margin of safety is not decreased by adding this reference to Section 6.9.A.6.b of the Quad Cities Technical Specifications and Bases Section 2.1.2 and Section 6.6.A.6.b of the LaSalle Technical Specifications. Siemens Power Corporation methodology for application of the ANFB Critical Power Correlation to coresident GE fuel is approved by the NRC and is the same methodology used in the cycle specific topical for coresident fuel (Reference 4 and 5). The MCPR Safety Limit will continue to ensure that greater than 99.9% of the rods in the core avoid boiling transition. Additionally, operating limits will be established to ensure the MCPR Safety Limit is not violated during all modes of operation.

Addition of SPC Topical for Revised ANFB Correlation Uncertainty (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

The MCPR Safety Limit provides a margin of safety by ensuring that less than 0.1% of the rods are expected to be in boiling transition if the MCPR Safety Limit is not violated. This Technical Specification amendment proposes to insert the topical report that describes SPC's calculation of the ATRIUM-9B additive constant uncertainty. The new ATRIUM-9B additive constant uncertainty calculation is conservative and is based on a larger database than previous calculations. Because a conservative method is used to calculate the ATRIUM-9B additive constant uncertainty, a decrease in the margin to safety will not occur due to adding this methodology to the Technical Specifications. In addition, operational limits will be established to ensure the MCPR Safety Limit is protected for all modes of operation. This revised methodology will only ensure that the appropriate level of fuel protection is being employed.

Change to Minimum Critical Power Ratio Safety Limit (Quad Cities Unit 1 and 2 and Dresden Units 2 and 3)

Changing the MCPR Safety Limit for Quad Cities and Dresden will not involve any reduction in margin of safety. The MCPR Safety Limit provides a margin of safety by ensuring that less than 0.1% of the rods are expected to be in boiling transition if the MCPR Safety Limit is not violated. The proposed Technical Specification amendment reflects the MCPR Safety Limit results from conservative evaluations by SPC using the

ANFB critical power correlation with the new 0.0195 ATRIUM-9B additive constant uncertainty documented in Reference 7 .

Because a conservative method is used to apply the ATRIUM-9B additive constant uncertainty in the MCPR Safety Limit calculation, a decrease in the margin to safety will not occur due to changing the MCPR Safety Limit. The revised MCPR Safety Limit will ensure the appropriate level of fuel protection. Additionally, operational limits will be established based on the proposed MCPR Safety Limit to ensure that the MCPR Safety Limit is not violated during all modes of operation including anticipated operation occurrences. This will ensure that the fuel design safety criterion of more than 99.9% of the fuel rods avoiding transition boiling during normal operation as well as during an anticipated operational occurrence is met.

Removal of Footnotes Limiting Operation with ATRIUM-9B Fuel Reloads (Quad Cities Unit 2 and Dresden Unit 3)

The removal of the cycle specific footnotes in Quad Cities and Dresden Technical Specifications does not impose a change in the margin of safety. These footnotes were added due to concerns regarding the calculation of the additive constant uncertainty for the ATRIUM-9B fuel and the cycle specific application of the ANFB critical power correlation to coresident GE fuel in Quad Cities Unit 2 Cycle 15. Because the generic ANFB application to coresident GE fuel MCPR methodology (Reference 3) has received NRC approval and the topical report describing the increased database used to calculate the additive constant uncertainties for ATRIUM-9B (Reference 7) have been submitted to the NRC and both are proposed to be added to the Technical Specifications in this amendment, there is no reason for the footnotes to remain. Removal of the Unit 2 specific "a" pages, 2-1a and B2-3a, in the Quad Cities Technical Specifications is justified by the removal of the footnotes. Therefore, the removal of the "a" pages, 2-1a and B2-3a, also does not impose a change in the margin of safety.

Revision to Thermal Limit Descriptions (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

The revision to the APLHGR and LHGR limit descriptions will not involve a reduction in the margin of safety. The methodology used to calculate the APLHGR must comply with the guidelines of Appendix K of 10 CFR Part 50, and the APLHGR and LHGR will still be required to be maintained within the limits specified in the COLR. The surveillance requirements for these two thermal limits remain unchanged. Thus, there will be no reduction in the margin of safety.

ATTACHMENT G

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

This proposed amendment does not involve a significant relaxation of the criteria used to establish the safety limits, a significant relaxation of the bases for the limiting safety system settings, or a significant relaxation of the bases for the limiting conditions for operations. Therefore, based on the guidance provided in 10CFR50.92(c), the proposed change does not constitute a significant hazards consideration.

Attachment H

Environmental Assessment Applicability Review

H. ENVIRONMENTAL ASSESSMENT APPLICABILITY REVIEW

ComEd has evaluated the proposed amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10CFR51.21. It has been determined that the proposed changes meet the criteria for categorical exclusion as provided for under 10 CFR 51.22(c)(9). This conclusion has been determined because the changes requested do not pose significant hazards considerations and do not involve a significant increase in the amounts, and no significant changes in the types of any effluents that may be released off-site. Additionally, this request does not involve a significant increase in individual or cumulative occupational radiation exposure.

Attachment I

References

ATTACHMENT I

REFERENCES

I. REFERENCES

1. ANF-91-048(P), Supplement 1, "BWR Jet Pump Model Revision for RELAX", Submitted to the NRC by SPC letter, "ANF-91-048(P), Supplement 1 and ANF-91-048(NP), Supplement 1, "BWR Jet Pump Model Revision for RELAX," RAC:96-042, R.A. Copeland to US NRC, May 6, 1996.

2. XN-NF-80-19(P), "Exxon Nuclear Methodology for Boiling Water Reactors -- Volume 2A, RELAX: A RELAP4 Based Computer Code for Calculating Blowdown Phenomena," June 1981.

3. EMF-1125(P)(A), Supplement 1 Appendix C, "ANFB Critical Power Correlation Application for Coresident Fuel", August 1997, and NRC SER, "Acceptance for Referencing of Licensing Topical Report EMF-1125(P), Supplement 1 Appendix C, 'ANFB Critical Power Correlation Application for Co-Resident Fuel", J. E. Lyons to R. A. Copeland, May 9, 1997.

4. EMF-96-021(P), Revision 1, "Application of the ANFB Critical Power Correlation to Coresident GE fuel for LaSalle Unit 2 Cycle 8", February 1996, and NRC SER, "Safety Evaluation for Topical Report EMF--95-021 (P), Revision 1, 'Application of the ANFB Critical Power Correlation to Coresident GE Fuel for LaSalle Unit 2 Cycle 8' (TAC NO. M94964)", D.M. Skay to I. Johnson, September 26,1996.

5. EMF-96-051(P), "Application of the ANFB Critical Power Correlation to Coresident GE Fuel for Quad Cities Unit 2 Cycle 15", May, 1996, and NRC SER, "Approval of Topical **Report EMF-96-051(P) - Quad Cities, Unit 2 (TAC NO. M96213)", R. Pulsifer to I. Johnson, May 16, 1997.**

6. ANF-1125(P)(A), Supplements 1 and 2, "ANFB Critical Power Correlation, Advanced Nuclear Fuels Corporation", April 1990.

7. ANF-1125(P), Supplement 1, Appendix D, "ANFB Critical Power Correlation Uncertainty For Limited Data Sets", Submitted to the NRC by SPC letter, "Request for Review of ANFB Critical Power Correlation Uncertainty for Limited Data Sets, ANF-1125(P), Supplement 1, Appendix D", HDC:97:032, H. D. Curet to Document Control Desk, April 18, 1997.

8. ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR ECCS Evaluation Model, January 1993.

9. "Dresden Nuclear Power Stations Units 2 and 3 Application for Amendment of Facility Operating License DPR-19 and DPR-25 Technical Specifications, NRC Docket Nos. 050-237 and 050-249," J.S. Perry to US NRC, June 20, 1996.

REFERENCES

10. "Dresden Nuclear Power Station Units 2 and 3 Supplement to Application for Amendment of Facility Operating Licenses DPR-19 and DPR-25 Technical Specifications", J.S. Perry to US NRC, December 30, 1996.

11. "Dresden Nuclear Power Station Units 2 and 3 Supplement to Application for Amendment of Facility Operating License DPR-19 and DPR-25 Technical Specifications", J.S. Perry to US NRC, March 5, 1997.

12. "LaSalle County Nuclear Power Station Units 1 and 2 Application for Amendment Request to Facility Operating Licenses NPF-11 and NPF-18, Technical Specifications Changes for Siemens Power Corporation Fuel Transition Docket Numbers 050-373 and 050-374", R.E. Querio to US NRC, April 8, 1996.

13. "LaSalle County Nuclear Power Station Units 1 and 2 Supplement to Application for Amendment of Facility Operating Licenses NPF-11 and NPF-18, Appendix A, Technical Specification Changes for Siemens Power Corporation Fuel Transition", W.T. Subalusky to U.S. NRC, October 14, 1996.

14. "Quad Cities Nuclear Power Stations Units 1 and 2, Application for Amendment Request to Facility Operating Licenses DPR-29 and DPR-30, Technical Specification Changes for Siemens Power Corporation (SPC) Fuel Transition, Docket Nos. 50-254 and 50-265", E.S. Kraft, to USNRC, June 10, 1996.

15. "Quad Cities Nuclear Power Stations Units 1 and 2 Supplement to Application for Amendment of Facility Operating License DPR-29 and DPR-30 Technical Specifications", E.S. Kraft to US NRC, February 17, 1997.

16. "Quad Cities Nuclear Power Station Units 1 and 2 Exigent Application for Amendment Request to Facility Operating Licenses Pursuant to 10CFR50.91(a)(6), DPR-29 and DPR-30, Technical Specification Changes for Revised Minimum Critical Power Ratio Safety Limit for Quad Cities Unit 2 Cycle 15, Docket Nos. 50-254 and 50-265", E.S. Kraft, Jr. to USNRC, April 21, 1997.

17. "Quad Cities Nuclear Power Station Units 1 and 2, Emergency Application for Amendment to Facility Operating Licenses Pursuant to 10CFR50.91, DPR-29 and DPR-30, Operation with ATRIUM-9B Fuel in Modes 3, 4, and 5", E.S. Kraft, Jr. to USNRC, April 29, 1997.