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ComEd

ESK-96-115

June 10, 1996

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

Subject: Quad Cities Nuclear Power Station 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 Numbers 50-254 and 50-265

Pursuant to 10 CFR 50.90, Commonwealth Edison (ComEd) proposes to amend Appendix A, Technical Specifications, of Facility Operating Licenses DPR-29 and DPR-30 to reflect the transition of fuel supplier from General Electric to SPC.

This proposed amendment request is subdivided as follows:

1. Attachment A provides a description and evaluation of the proposed changes in this amendment.

2. Attachment B includes a summary of the proposed changes, index of the changes, and the marked up pages of the Technical Specifications.

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Change NRC PDR 1 ltr and 1 INP

3. Attachment C describes ComEd's evaluation performed in accordance with 10 CFR 50.92(c), which confirms that no significant hazard consideration is involved.

4. Attachment D provides the Environmental Assessment Applicability Review.

5. Attachment E is an SPC Licensing Methodology Summary document for Boiling Water Reactors (EMF-94-217), which has been prepared to provide an integrated summary of the various approved topical reports, design criteria and licensing methods used by SPC. This document is provided to facilitate the staff's review of this amendment and does not contain any new or revised SPC methods.

6. Attachment F is an affidavit to support withholding Attachment E from public disclosure.

Generic licensing topical reports for SPC Boiling Water Reactor (BWR) methodologies have been approved by the Nuclear Regulatory Commission (NRC) and have been applied at other BWR plants (e.g., Susquehanna, Washington Nuclear Power Unit 2, Dresden, Grand Gulf). The ATRIUM-9B fuel design, which will be used in the initial Quad Cities reloads, is a NRC approved design.

Two other separate licensing actions are discussed here for completeness and do not affect the content of the proposed Technical Specifications: (1) the Critical Power Ratio (CPR) treatment of the mixed core and (2) Jet Pump modeling used by Siemens in their Loss of Coolant Accident (LOCA) analysis. A generic methodology document for the CPR treatment of the mixed core (EMF-1125(P) Supplement 1, Appendix C, "ANFB Critical Power Correlation Application for Co-Resident Fuel") has been submitted to the Staff (Reference: Letter dated November 30, 1995, Siemens Power Corporation to NRR, "Submittal of EMF-1125(P), Supplement 1 Appendix C"). Approval of the generic document is not anticipated prior to startup of Unit 2 Cycle 15, therefore, a cycle specific document for the CPR treatment of the mixed core is being prepared and will be submitted for approval in support of Unit 2 Cycle 15 (expected to be sent June, 1996). The SPC LOCA analysis methodology contains a significant conservatism for small-break LOCAs, that impacts maximum Average Planar Linear Heat Generation Rate (APLHGR) values. Supplement 1 to ANF-91-048 was submitted by Siemens on May 6, 1996. Approval is anticipated prior to Unit 2 Cycle 15 startup.

As discussed previously with the NRC in a conference call (September 15, 1994) and related letter (G. Benes to US Nuclear Regulatory Commission, "LaSalle County Nuclear Power Station Units 1 and 2 Quad Cities Nuclear Power Station

Units 1 and 2 Fuel Vendor Transition for Post-1995 Reloads," November 14, 1994), the Minimum Critical Power Ratio (MCPR) Safety Limit will be validated by cycle-specific analyses by SPC.

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

The following is an overview of the changes:

> Quad Cities Nuclear Power Station currently operates with General Electric (GE) fuel and methodologies. ComEd performs the core designs using NRC approved methodologies. SPC will provide fuel and related support services for Quad Cities beginning with Unit 2 Cycle 15 and Unit 1 Cycle 16. The fuel assembly designs and methods used by SPC are NRC approved for Boiling Water Reactors (See Attachment E, EMF-94-217(P)). Other BWR plants have been licensed and operated with SPC fuel. Since the changes deal with a transition from one set of NRC approved methods to another, the amendment is largely administrative. The majority of the changes being made are in the Bases of the Technical Specifications and the methodologies reference list in Technical Specifications Section 6.

As mentioned above, Siemens Power Corporation considers some of the information contained in EMF-94-217(P), Revision 1, "Boiling Water Reactor Licensing Methodology Summary," to be proprietary. In accordance with the requirements of 10 CFR 2.790(b), an affidavit is enclosed to support withholding of this document from public disclosure. A non-proprietary version is also provided.

This amendment is needed to support Quad Cities operations with Siemens fuel. Siemens fuel will be initially loaded into Quad Cities Unit 2 Cycle 15 (startup approximately March, 1997) and Quad Cities Unit 1 Cycle 16 (startup approximately March, 1998); therefore, this amendment is required prior to the startup of Unit 2 Cycle 15. It is requested that the amendment be approved by March 1, 1997, with implementation prior to the startup of Cycle 15 for Unit 2. The changes being made eliminate the need for a separate amendment for Unit 1 Cycle 16.

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

Commonwealth Edison is notifying the Illinois Department of Nuclear Safety (IDNS) of this application for amendment by transmitting a copy of this letter and its attachments to the designated state official.

Please direct any questions or comments concerning this letter to Nick Chrissotimos, Regulatory Assurance, at (309) 654-2241, extension 3100.

Sincerely,

E. S. Kraft, Jr.
Site Vice President
Quad Cities Nuclear Power Station

State of Illinois
County of Rock Island
Signed before me on June 10, 1996
by Edward S. Kraft, Jr.

Linda L. Stoermer, Notary Public

"OFFICIAL SEAL"
LINDA LEE STOERMER
Notary Public, State of Illinois
My Commission Expires 11/19/96

Attachments:

(A) Description and Evaluation of the Proposed Changes
(B) Summary of Proposed Changes, including Marked-up Technical Specification Pages
(C) Evaluation of Significant Hazards Considerations
(D) Environmental Assessment Applicability Review
(E) Boiling Water Reactor Licensing Methodology Summary, Siemens Power Corporation, EMF-94-217(P)
(F) Proprietary Withholding Affidavit for EMF-94-217(P)

cc: H. J. Miller - Regional Administrator, Region III
R. M. Pulsifer - Quad Cities NRC Project Manager
C. G. Miller - Quad Cities Senior Resident Inspector
Office of Nuclear Facility Safety - IDNS
D. C. Tubbs, MidAmerican Energy Co.
R. J. Singer, MidAmerican Energy Co.

Attachment F

SPC affidavit Supporting Non-disclosure of EMF-94-217

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

I, R. A. Copeland being duly sworn, hereby say and depose:

1. I am Manager, Product Licensing, for Siemens Power Corporation ("SPC"), and as such I am authorized to execute this Affidavit.

2. I am familiar with SPC's detailed document control system and policies which govern the protection and control of information.

3. I am familiar with the topical report EMF-94-217(P), Revision 1, entitled "Boiling Water Reactor Licensing Methodology Summary," referred to as "Document." Information contained in this Document has been classified by SPC as proprietary in accordance with the control system and policies established by SPC for the control and protection of information.

4. The Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by SPC and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in the Document as proprietary and confidential.

5. The Document has been made available to the U.S. Nuclear Regulatory Commission in confidence, with the request that the information contained in the Document will not be disclosed or divulged.

6. The Document contains information which is vital to a competitive advantage of SPC and would be helpful to competitors of SPC when competing with SPC.

7. The information contained in the Document is considered to be proprietary by SPC because it reveals certain distinguishing aspects of SPC licensing methodology which secure competitive advantage to SPC for fuel design optimization and marketability, and includes information utilized by SPC in its business which affords SPC an opportunity to obtain a competitive advantage over its competitors who do not or may not know or use the information contained in the Document.

8. The disclosure of the proprietary information contained in the Document to a competitor would permit the competitor to reduce its expenditure of money and manpower and to improve its competitive position by giving it valuable insights into SPC licensing methodology and would result in substantial harm to the competitive position of SPC.

9. The Document contains proprietary information which is held in confidence by SPC and is not available in public sources.

10. In accordance with SPC's policies governing the protection and control of information, proprietary information contained in the Document has been made available, on a limited basis, to others outside SPC only as required and under suitable agreement providing for nondisclosure and limited use of the information.

11. SPC policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

12. Information in this Document provides insight into SPC licensing methodology developed by SPC. SPC has invested significant resources in developing the methodology as well as the strategy for this application. Assuming a competitor had available the same background data and incentives as SPC, the competitor might, at a minimum, develop the information for the same expenditure of manpower and money as SPC.

THAT the statements made hereinabove are, to the best of my knowledge, information, and belief, truthful and complete.

FURTHER AFFIANT SAYETH NOT.

SUBSCRIBED before me this 13th

day of November, 1995.

Susan K. McCoy
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 1/10/96

Attachment A

Description and Evaluation of the Proposed Changes

A. Description and Evaluation of the Proposed Changes

1. Background Information

Quad Cities Nuclear Power Station operates with General Electric (GE) fuel and methodologies. ComEd performs core designs using NRC approved methodologies. Siemens Power Corporation (SPC) will provide fuel and related support services for Quad Cities beginning with Unit 2 Cycle 15 and Unit 1 Cycle 16. The fuel designs and methods used by SPC are Nuclear Regulatory Commission (NRC) approved for Boiling Water Reactors (See Attachment E, EMF-94-217(P)). Other Boiling Water Reactor (BWR) plants (e.g., Grand Gulf, Washington Nuclear, Dresden, Susquehanna) have been licensed and operated with SPC fuel. The following is a discussion of the topics in the Technical Specifications affected by this change in fuel vendors. Since the changes deal with a transition from one set of NRC approved methods to another, the amendment requested is largely administrative. The majority of the changes being made are in the Bases of the Technical Specifications and the methodologies reference list in the Technical Specifications. For details of the individual changes, see the discussion below for the particular change.

Power Distribution Limits:

The LINEAR HEAT GENERATION RATE (LHGR) limit is defined as the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length. LHGR is calculated and monitored in units of kilowatt per foot. Excessive LHGR values (high kW/ft) can cause the fuel pellet to expand to the point of overstressing the cladding. Operating the fuel within its design LHGR limits, combined with analyses of Anticipated Operational Occurrences (AOOs), ensures that 1% plastic strain of the cladding is not exceeded.

The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) limit is applicable to a specific planar height and is equal to the sum of the LHGRs for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle. Operating the fuel within its APLHGR limits ensures that 10CFR50.46 acceptance criteria are met during a loss of coolant accident (LOCA).

The MINIMUM CRITICAL POWER RATIO (MCPR) limit is the minimum CRITICAL POWER RATIO (CPR) which exists in the core, where CPR is the ratio of that power in the assembly which is calculated by application of an NRC approved correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power. Operating a bundle at a power level beyond that required for the onset of transition boiling creates a condition with poor heat transfer and may cause fuel failure due to the elevated cladding temperature. Operating limits for MCPR are set so that the MCPR Safety Limit is not exceeded during Anticipated Operational Occurrences. The Safety Limit is set for MCPR such that 99.9% of the fuel rods avoid boiling transition if the Safety Limit is not violated. Core transients are analyzed for Anticipated Operational Occurrences to determine the change in MCPR during the transients. MCPR is calculated using the critical power correlation of record. The MCPR Safety Limit is listed as 1.07 (1.08 for SLO) on page 2-1. SPC methods validate the MCPR Safety Limit each operating cycle. This is a routine part of SPC analyses, which includes an evaluation of the effects of channel bow in determining the MCPR Safety Limit (Reference: "ANFB Critical Power Correlation," ANF-1125(P)(A)). The value is projected to remain 1.07 (1.08 for SLO). If the cycle specific SPC analyses, however, show it to require a different value, a supplemental transmittal will be submitted. If needed, the supplemental submittal would be in the September 1996 time frame. This is consistent with previous discussions held between ComEd and the NRC.

GE LHGR and APLHGR limits will be applied to the co-resident GE fuel in the core, while SPC LHGR and APLHGR limits will be applied to the SPC fuel in the core. The LHGR Basis in the existing Technical Specifications applies to both GE and SPC fuel. The Technical Specifications Basis for the GE APLHGR will remain and the SPC Basis for APLHGR will be added. The GE fuel LHGR is monitored via Maximum Fraction of Limiting Power Density (MFLPD). The SPC fuel LHGR is monitored via Fuel Design Limiting Ratio (FDLRX) and Fuel Design Limiting Ratio for Centerline Melt (FDLRC). The fuel LHGR is limited during transients from off rated conditions by the application of an Average Power Range Monitor (APRM) setpoint requirement when the power shape is strongly peaked. The GE fuel LHGR limit is protected by adjusting the setpoint (or APRM readings) when the MFLPD is greater than the Fraction of Rated Power (FRP). For SPC fuel, the setpoint requirement is applied based on a SPC provided transient LHGR limit, implemented by the parameter FDLRC.

The SPC critical power correlation, ANFB, will be the licensed correlation for the reload and will be used to analyze core transients for MCPR protection. The MCPR of co-resident GE fuel will be calculated using bundle geometry dependent constants so the mean of the ANFB calculated CPR data is conservative relative to that calculated by the GE correlation (GEXL). This information does not affect the content of the proposed changes to the Technical Specifications, but is being included for completeness.

The scram timing terminology in Bases 3/4.3.F and 3/4.11.C are modified to reflect the methods used, by referring to the nominal times used in the Core Operating Limits Report (COLR) as well as the existing Technical Specifications required times. The changes being proposed involve one Technical Specifications amendment; for Unit 1 Cycle 15, the COLR will address the use of the nominal scram speeds with the GE methods being utilized.

Miscellaneous Changes:

The Reactivity Anomaly surveillance is performed to check actual core reactivity against predicted core reactivity. The current Specification refers to a comparison of control rod density. The specification and its basis are modified to enable use of the k_{eff} method for monitoring core reactivity as well as the control rod density. The Specification will allow either method to be used to provide an alternate backup method.

Since the SPC ATRIUM-9B fuel contains a water box, the wording of the fuel description (page 5-5) is modified to reflect this.

The pressure, temperature and volume details of the reactor coolant system in Specification 5.4 are being re-located to the UFSAR as a line item from Improved Technical Specifications (Reference: NUREG-1434).

The peak power level during the low power Rod Withdrawal Error (RWE) event noted on page B 2-5 is revised from 1% to 7.7%. This number refers to the power level at which the IRM system terminates the low power RWE event. The currently listed value of 1% relates the core average power level during the event. The revised number of 7.7% power is the equivalent number for the local bundle power as described in UFSAR section 7.6.1. Both numbers are correct for the parameter they reference. However, the local power level is more relevant to the MCPR protection discussed in the Technical Specification bases.

The Bases discussion (pages B 3/4.3-6 and B 3/4.3-7) for the Rod Worth Minimizer was clarified to state that the control rod sequences used during the cycle are not all written prior to cycle startup, but are verified to meet the 280 cal/gm limit up to the Low Power Set Point (LPSP) using NRC approved methodology. This methodology is referenced in Technical Specification 6.9.A.6, but is not described in detail.

The Bases discussion (page B 3/4.6-3) of pressurization transients for the ASME over-pressurization event is modified to reflect the fact that Siemens' methodology determines the most limiting pressurization transient each fuel cycle. The Main Steam Line Isolation Valve (MSIV) closure with flux scram may not necessarily be the limiting pressurization event.

The references in section 6 are modified to include SPC references.

2. Power Distribution Limits

a. **Linear Heat Generation Rate (LHGR)**:

Affected pages: I, II, XIII, XXVI, 1-3, 1-7, 3/4.11-2, B 3/4.11-1

Description of the current requirements (LHGR)

The LHGR is the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length. LHGR is calculated and monitored in units of kilowatt per foot. General Electric fuel has fuel specific LHGR limits and compliance with this limit is monitored by the parameters FRACTION OF LIMITING POWER DENSITY(FLPD) and MAXIMUM FRACTION OF LIMITING POWER DENSITY(MFLPD). Technical Specification 3/4.11.D requires that the fuel be operated with an LHGR that does not exceed the LHGR limit specified in the Core Operating Limits Report (COLR). The definitions of LHGR, FLPD and MFLPD are on pages I and 1-3 of the Technical Specifications.

The current Technical Specifications (3/4.11.B) require the flow-referenced Average Power Range Monitor (APRM) trips to be lowered or the APRM readings to be increased when the MFLPD exceeds the Fraction of Rated Thermal Power (FRTP).

Basis for the current requirements (LHGR)

One of the design limits for GE fuel is the fuel type specific LHGR limit, monitored as a Fraction of Limiting Power Density (FLPD) and Maximum Fraction of Limiting Power Density (MFLPD). FLPD and MFLPD are the ratio of the LHGR to its limit for a bundle and the maximum ratio for any bundle in the core, respectively. The LHGR limit, combined with analyses of abnormal operational occurrences, ensures that 1% plastic strain of the cladding is not exceeded. The effects of fuel densification are accounted for in the GE fuel design methods.

With MFLPD higher than FRTP, there is a relatively large amount of power shape peaking in the core. Since a flow-referenced scram would not occur as quickly as it should under the high peaking conditions, the fuel could exceed the 1% plastic strain design limits if a power increase transient were to occur. Lowering the APRM flow-referenced trips provides assurance that a scram will occur prior to the 1% plastic strain design limit being exceeded.

Need for revision of the requirements (LHGR)

The subject LHGR terminology (FLPD and MFLPD) is specific to GE. Since ComEd is beginning a transition from GE to Siemens Power Corporation (SPC) fuel and core monitoring methods, the Technical Specifications Bases need to be modified to include the SPC terminology for LHGR. The co-resident GE fuel in the core will be monitored via the GE fuel dependent LHGR limits and the SPC fuel will be monitored via SPC LHGR limits. Technical Specification 3/4.11.D remains unchanged. The plant will still be required to maintain the LHGRs less than or equal to the LHGR limit specified in the Core Operating Limits Report (COLR).

SPC fuel is protected from transients from off rated conditions by the application of the FUEL DESIGN LIMITING RATIO FOR CENTERLINE MELT (FDLRC) to the APRM setpoints. This application is needed for the SPC fuel.

Description of the revised requirements (LHGR)

The definitions of the GE LHGR limits (FLPD and MFLPD) are annotated to indicate they apply to GE fuel. Affected pages are I and 1-3.

The SPC LHGR limits are the FUEL DESIGN LIMITING RATIO (FDLRX) and FUEL DESIGN LIMITING RATIO FOR CENTERLINE MELT (FDLRC). The definitions for these SPC LHGR limits are added to the index on page I of the Technical Specifications. These definitions are also added to page 1-3, as follows:

FUEL DESIGN LIMITING RATIO (FDLRX)

The FUEL DESIGN LIMITING RATIO (FDLRX) shall be the limit to assure that the fuel operates within the end-of-life steady-state design criteria by, among other items, limiting the release of fission gas to the cladding plenum (applicable to SPC Fuel).

FUEL DESIGN LIMITING RATIO FOR CENTERLINE MELT (FDLRC)

The FUEL DESIGN LIMITING RATIO FOR CENTERLINE MELT (FDLRC) shall be 1.2 times the LHGR at a given location divided by the product of the TRANSIENT LINEAR HEAT GENERATION RATE limit and the FRACTION OF RATED THERMAL POWER (applicable to SPC Fuel).

The definition of the SPC transient LHGR limit is also added to the index on p. I of the Technical Specifications. It is added to page 1-7 as follows:

TRANSIENT LINEAR HEAT GENERATION RATE (TLHGR)

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

The APRM setpoints in Specifications 3/4.11.B are modified to reflect the SPC FDLRC limit and the requirement to modify the APRM setpoints if FDLRC is greater than 1.0 for SPC fuel. Index pages XIII and XXVI are updated to reflect the modified section. A footnote is added to address GE fuel; the same monitoring requirements and actions apply based on MFLPD/FRP in the place of FDLRC. This makes the SPC FDLRC Technical Specification consistent with that of Dresden, while minimizing the need for future License amendments. The Basis for the APRM setpoints (page B 3/4.11-1) has the first sentence deleted. This is because SPC determines the MCPR Safety Limit based on the limiting power distribution in the cycle-specific step-through (i.e., the power distribution that yields the most limiting result for the MCPR fuel cladding safety limit may not necessarily yield the design LHGR at rated power). The basis for the APRM setpoint adjustment is supplemented with the following:

SPC Fuel

The Fuel Design Limiting Ratio for Centerline Melt (FDLRC) is incorporated to protect the above criteria at all power levels considering events which cause the reactor power to increase to 120% of rated thermal power.

The scram settings must be adjusted to ensure that the TRANSIENT LINEAR HEAT GENERATION RATE (TLHGR) is not violated for any power distribution. This is accomplished using FDLRC. The scram setting is decreased in accordance with the formula in Specification 3.11.B [see definition of TLHGR discussed above], when FDLRC is greater than 1.0.

The adjustment may also be accomplished by increasing the gain of the APRM by FDLRC. This provides the same degree of protection as reducing the trip setting by 1/FDLRC by raising the initial APRM reading closer to the trip setting such that a scram would be received at the same point in a transient as if the trip setting had been reduced.

Basis for the revised requirements (LHGR)

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 mechanism protected 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.

The Fuel Design Limiting Ratio for Centerline Melt (FDLRC) is incorporated to protect the above criteria at all power levels considering events which cause the reactor power to increase to 120% of rated thermal power.

The scram settings must be adjusted to ensure that the LHGR transient limit (TLHGR) is not violated for any power distribution. This is accomplished using FDLRC. The scram setting is decreased in accordance with the formula in Specification 3.11.B, when FDLRC (or MFLPD/FRP) is greater than 1.0.

The adjustment may also be accomplished by increasing the APRM gain by FDLRC. This provides the same degree of protection as reducing the trip setting by 1/FDLRC (or FRP/MFLPD) by raising the initial APRM reading closer to the trip setting such that a scram would be received at the same point in a transient as if the trip setting had been reduced.

Since the core will contain co-resident GE fuel, the GE LHGR limits are being retained. The GE fuel will be monitored using GE LHGR limits and the SPC fuel will be monitored using SPC LHGR limits. The SPC basis for LHGR is the mechanical integrity of the fuel and includes the design criteria of less than one percent plastic strain and avoidance of fuel centerline melt.

The SER for ANF-89-014, Advanced Nuclear Fuels Corporation "Generic Mechanical Design for Advanced Nuclear Fuels 9x9-IX and 9x9-9X BWR Reload Fuel," discusses the mechanical design analyses performed by SPC for the subject fuel designs. It concludes that the 9x9-IX (currently known as ATRIUM-9B) and 9x9-9X designs as described in ANF-89-014 are acceptable for licensing applications for BWRs, with the exception that plant-specific analysis of a seismic/LOCA event is required for reload applications. The SPC analysis of a Seismic-LOCA event is in progress as part of the Unit 2 Cycle 15 reload licensing calculations. Attachment E (EMF-94-217(P), Boiling Water Reactor Licensing Methodology Summary) summarizes the Siemens methods and refers to the NRC approved documents regarding the methods used.

Specific calculations are performed to ensure a 1% plastic strain design requirement is met for the reload core. For the time period in which both GE fuel and SPC fuel reside in the core, GE will provide information to ComEd that will allow the continued compliance with the 1% plastic strain design requirement for the GE fuel types that remain in the core.

b. Average Planar Linear Heat Generation Rate (APLHGR):

Affected page: B 3/4.11-1

Description of the current requirements (APLHGR)

The APLHGR is 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. Technical Specification 3/4.11.A requires that the fuel be operated with an APLHGR that does not exceed the APLHGR limits specified in the COLR.

Basis for the current requirements (APLHGR)

The APLHGR specification for the GE fuel assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10CFR50.46. This specification also assures that fuel rod mechanical integrity is maintained during normal and transient operations.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. The calculational procedure used to establish the maximum APLHGR values uses NRC approved calculational models which are consistent with the requirements of 10CFR50.46. The approved calculational models are listed in Specification 6.9.

Need for revision of the requirements (APLHGR)

Both GE and SPC use APLHGR to protect the fuel cladding from exceeding 10CFR50.46 limits during the design basis LOCA. GE uses APLHGR to ensure mechanical integrity of the fuel, while SPC uses LHGR for this purpose. Therefore, the Bases for the GE fuel is being retained and the Bases for the SPC fuel is being added. Each fuel type (GE and SPC) will be monitored using its respective vendor supplied APLHGR limits. Technical Specification 3/4.11.A remains unchanged. The plant will still be required to maintain the APLHGR less than or equal to its limits as specified in the COLR.

Description of the revised requirements (APLHGR)

The basis for the GE fuel APLHGR is modified on page B 3/4.11-1 to annotate the generic Bases, those bases as being applicable to GE fuel, and a new section has been added which discusses the basis for the SPC application of the APLHGR limits, as follows:

SPC Fuel

This specification assures that the peak cladding temperature of SPC fuel following a postulated design basis loss-of-coolant accident will not exceed the Peak Cladding Temperature (PCT) and maximum oxidation limits specified in 10CFR50.46. The calculational procedure used to establish the Average Planar Linear Heat Generation Rate (APLHGR) limits is based on a loss-of-coolant accident analysis.

The PCT following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod-to-rod power distribution within the assembly.

The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits for two-loop and single-loop operation are specified in the Core Operating Limits Report (COLR).

Basis for the revised requirements (APLHGR)

APLHGR is monitored for GE fuel to ensure mechanical integrity of the fuel rods and to maintain the peak clad temperature during the Design Basis Loss of Coolant Accident less than the 10CFR50.46 limit. APLHGR is monitored for SPC fuel to limit peak clad temperature while the mechanical integrity of the fuel is maintained via LHGR monitoring. These differences require both Bases to be included in the Technical Specifications. The GE fuel will be monitored using GE APLHGR limits and the SPC fuel will be monitored using SPC APLHGR limits; therefore, a separate amendment will not be required for Unit 1. These limits will be identified in the COLR. The SER for [illegible]-89-014 (P)(A), Advanced Nuclear Fuels Corporation, "Generic Mechanical Design for Advanced Nuclear Fuels 9x9-IX [ATRIUM-9B] and 9x9-9X BWR Reload Fuel," states: "The ANF design criteria for ECCS evaluation met the requirements of 10 CFR 50.46 as it relates to cladding embrittlement for a LOCA; i.e., the criteria of a peak cladding temperature limit of 2200 degrees Fahrenheit and a 17% limit on maximum cladding oxidation.

We conclude that these criteria or limits are also applicable for application to the 9x9-IX and 9x9-9X designs up to the burnup levels requested in ANF-89-014.
Evaluation - The principal cause of cladding embrittlement during severe accidents such as LOCA is the high cladding temperatures that result in severe cladding oxidation. The ANF methodology for evaluating cladding oxidation and embrittlement during a LOCA is included in their approved report for LOCA-ECCS analysis," EXEM-ECCS Evaluation, XN-NF-80-19(P)(A), Volumes 2, 2A, 2B, and 2C.

Further discussion of the SPC LOCA-ECCS methods can be found in ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR ECCS Evaluation Model," and EMF-94-217(P), "Boiling Water Reactor Licensing Methodology Summary."

c. Critical Power Ratio (CPR):

Affected pages: B 3/4.3-3, B 3/4.3-4, B 3/4.11-2, 6-15

Description of the current requirements (CPR)

The MCPR is the smallest CPR which exists in the core, where CPR (CRITICAL POWER RATIO) is the ratio of that power in the assembly which is calculated by application of the NRC approved correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power. The operating CPR needs to be monitored and maintained above the operating limit during normal operation to ensure that the Safety Limit will not be exceeded during Anticipated Operational Occurrences (AOOs), should they occur. The Safety Limit is set such that 99.9% of the fuel rods avoid boiling transition if the Safety Limit is not violated. Technical Specification 3/4.11.C requires that the fuel be operated with the Minimum CPR greater than or equal to the MCPR limit specified in the COLR. Technical Specification 3/4.3.E specifies the time requirements for control rod scram insertion.

Basis for the current requirements (CPR)

The current requirements are based on GE methods, including the use of the GE CPR correlation (GEXL) to calculate the CPR for the GE fuel bundles. GE methods are also used to determine the fuel MCPR Safety Limit. GE transient analysis methods are used to determine the delta CPR for various Anticipated Operational Occurrences (AOOs). As stated in the description above, the basis for the CPR is to avoid boiling transition in the fuel bundle.

The required operating limit for MCPR at normal operating conditions is derived from the established fuel cladding integrity Safety Limit MCPR and an analysis of abnormal operational transients. For any AOO transient analysis evaluation with the initial condition of the reactor being at the steady-state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming Reactor Protection System instrument trip settings given in Technical Specification 2.2.

To ensure that the fuel cladding integrity Safety Limit is not exceeded during any AOO transient, the most limiting transients are analyzed to determine which event results in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated are change of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR is obtained and presented in the CORE OPERATING LIMITS REPORT (COLR). This operating limit may be a function of control rod scram times. Flow dependent operating limits are provided to decrease the operating limit for MCPR at off rated conditions. This assures that the Safety Limit will not be exceeded during transients initiated from off rated conditions.

Need for revision of the requirements (CPR)

With changes from GE to SPC fuel and methods, it is necessary to update the Technical Specifications Bases to accommodate the SPC methods for assuring margin to transition boiling. When a Unit installs SPC fuel, the GE critical power correlation (GEXL) will no longer be used; instead, the SPC critical power correlation (ANFB) will be the correlation of record. As discussed below (basis for the revised requirements), the GE fuel will be monitored using the ANFB correlation with additive constants that are used to ensure the ANFB results are conservative.

Technical Specification 3/4.11.C remains unchanged. The plant will still be required to maintain the MCPR greater than or equal to its operating limit as specified in the COLR. The operating limit evaluations for MCPR include scram time dependence. GE methods utilize the 20% scram insertion point to determine the operating limit for MCPR. The measured scram times are used by both vendor methodologies to determine the operating limit for MCPR using analyses based on the Technical Specification times and analyses based on nominal scram times, as specified in the COLR. GE methods utilize recent as well as past data from the current cycle in determining the operating limit for MCPR.

SPC methods evaluate the 5%, 50%, and 90% insertion times in addition to the 20% time. The most recent data for each control rod is used to determine the operating limit for MCPR. If the scram times do not meet the nominal times, the operating limit associated with the Technical Specification times is used. Changes are being made to the Bases for 3/4.3 and 3/4.11.C to clarify the methods being used. As discussed below, the changes to the Specifications and Bases support the change without need for a separate amendment for Unit 1 Cycle 16.

Description of the revised requirements (CPR)

The reference to adjustment of the scram requirements for past data is deleted on page B 3/4.3-3 and B 3/4.3-4. The text on page B 3/4.3-4 discussing the GE methods of determining the operating limit for MCPR, as a result of scram timing is replaced by the following:

> Transient analyses are performed for both Technical Specification Scram Speed (TSSS) and nominal scram speed (NSS) insertion times. These analyses result in the establishment of the cycle dependent TSSS MCPR limits and NSS MCPR limits presented in the COLR. Results of the control rod scram tests performed during the current cycle are used to determine the operating limit for MCPR. Following completion of each set of scram testing, the results will be compared with the assumptions used in the transient analysis to verify the applicability of the MCPR operating limits. Prior to the initial scram time testing for an operating cycle, the MCPR operating limits will be based on the TSSS insertion times.

The detailed reference to 20% scram insertion time is deleted from statement 4 of item 6 on page 6-15.

The text of the MCPR Basis on page B 3/4.11-2 is modified so that "have been " and "were" are each changed to "are." This is to reflect the transients being analyzed each cycle. The Bases regarding the purpose of the flow biasing of the MCPR operating limit is replaced by the following:

> MCPR Operating Limits are presented in the CORE OPERATING LIMITS REPORT (COLR) for both Nominal Scram Speed (NSS) and Technical Specification Scram Speed (TSSS) insertion times. The negative reactivity insertion rate resulting from the scram plays a major role in providing the required protection against violating the Safety Limit MCPR during transient events. Faster scram insertion times provide greater protection and allow for improved MCPR performance. The application of NSS MCPR limits utilizes measured data that is faster than the times required by the Technical Specifications, while the TSSS MCPR limits provide the necessary protection for the slowest allowable average scram insertion times identified in Specification 3.3.E. The measured scram times are compared with the nominal scram insertion times and the Technical Specification Scram Speeds. The appropriate operating limit is applied, as specified in the COLR.
>
> For core flows less than rated, the MCPR Operating Limit established in the specification is adjusted to provide protection of the Safety Limit MCPR in the event of an uncontrolled recirculation flow increase to the physical limit of the pump. Protection is provided for manual and automatic flow control by applying the appropriate flow dependent MCPR limits presented in the COLR. The MCPR Operating Limit for a given power/flow state is the greater value of MCPR as given by the rated conditions MCPR limit or the flow dependent MCPR limit. For automatic flow control, in addition to protecting the Safety Limit MCPR during the flow run-up event, protection is provided to prevent exceeding the rated flow MCPR Operating Limit during an automatic flow increase to rated core flow.

Basis for the revised requirements (CPR)

The discussion of scram time dependent methods for determining the MCPR operating limit are modified to clarify the use of nominal and Technical Specification times and accommodate both vendors' methods. The negative reactivity insertion rate resulting from the scram plays a major role in providing the required protection against violating the Safety Limit MCPR during transient events. Faster scram insertion times provide greater protection and allow for improved MCPR performance. The application of NSS MCPR limits utilizes measured data that is faster than the times required by the Technical Specifications, while the TSSS MCPR limits provide the necessary protection for the slowest allowable average scram insertion times identified in Specification 3/4.3.E. The measured scram times are compared with the nominal scram insertion times and the Technical Specification Scram Speeds. The appropriate operating limit is applied, as specified in the COLR. This process of using measured scram insertion times for determining the MCPR Operating Limit is a standard BWR method used with both General Electric and Siemens methodology. The discussion of the process for determining the MCPR operating limit in the Bases on pages B 3/4.3-3 and B 3/4.3-4 are modified to include clarification of the use of nominal times being maintained in the COLR.

The Basis for MCPR on page B 3/4.11-2 is also modified to reflect the SPC application of flow dependent MCPR operating limits at non-rated flows to protect against transients initiated from off rated conditions.

The NRC approved SPC CPR correlation (ANFB) is documented in ANF/EMF-1125, ANFB Critical Power Correlation. Each vendor has developed its own proprietary correlation for determining the fuel assembly critical power. Since future reload fuel will be supplied by SPC, the SPC ANFB correlation will be used for determining the critical power in the mixed cores. As such, the critical power for the SPC fuel will be determined with the ANFB correlation. The critical power for the existing co-resident GE fuel will also be based on the ANFB correlation. However, for the GE fuel, appropriate bundle geometry constants will be used with the ANFB critical power correlation to ensure that the mean of the ANFB calculated critical power results for the GE fuel is conservative relative to the results that would be determined with the GE GEXL correlation. A generic methodology document for the CPR treatment of the mixed core (EMF-1125(P) Supplement 1, Appendix C, "ANFB Critical Power Correlation Application for Co-Resident Fuel"), has been submitted to the Staff (Reference: Letter dated November 30, 1995, Siemens Power Corporation to NRR, "Submittal of EMF-1125(P), Supplement 1 Appendix C"). It is anticipated that EMF-1125 Supplement 1 Appendix C, "ANFB Critical Power Correlation Application for Co-Resident Fuel," will be generically approved by the NRC prior to the first SPC cycle for Unit 1 (late 1997). However, approval of the generic document is not anticipated prior to startup of Unit 2 Cycle 15. Therefore, a cycle specific document for the CPR treatment of the mixed core will be prepared and submitted for approval in support of Unit 2 Cycle 15 (EMF-96-051(P), "Application of the ANFB Critical Power Correlation to Coresident GE Fuel for Quad Cities Unit 2 Cycle 15," May 1996).

3. Miscellaneous Changes

a. Reactivity Anomaly Surveillance:

Affected pages: II, 1-6, 3/4.3-2, B 3/4.3-2

Description of the current requirements (Reactivity Anomaly)

Technical Specification 3/4.3.B requires that the reactivity equivalence of the difference between the actual ROD DENSITY and the predicted ROD DENSITY be less than or equal to 1% delta k/k.

Basis for the current requirements (Reactivity Anomaly)

Per Standard Technical Specifications, the reactivity anomaly limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between monitored and predicted core reactivity may indicate that the assumptions of the Design Basis Accident (DBA) and transient analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. A limit on the difference between the monitored reactivity and the predicted reactivity of 1% delta k/k has been established. A deviation greater than 1% from that predicted is larger than expected for normal operation and should therefore be evaluated.

Need for revision of the requirements (Reactivity Anomaly)

The SPC core monitoring code (POWERPLEX), as well as the existing GE Core Monitoring Code (CMC), enables the site to monitor predicted Keff vs. actual Keff. In order to use this capability, the reference to ROD DENSITY is being deleted and critical control rod configuration needs to be added. The Bases is also modified to include discussion of the Keff method.

Description of the revised requirements (Reactivity Anomaly)

The phrase ROD DENSITY is deleted from the definitions in the Specifications (page II). The phrase ROD DENSITY is changed to "critical control rod configuration" on page 3/4.3-2 to enable either method to be used. The following is added to the Bases for 3/4.3.B:

> Alternatively, monitored Keff can be compared with the predicted Keff as calculated by the 3D core simulator code.

Basis for the revised requirements (Reactivity Anomaly)

The change enables the site to use the Keff method of monitoring for reactivity anomalies that is available with the POWERPLEX monitoring code. This method is being used currently at Dresden. Monitoring core reactivity via control rod density utilizes a correlation between a change in control rod density and core reactivity. The method of using k_{eff} is a more direct measurement method and is consistent with NUREG-1434. The capability to use control rod configuration is retained as an alternate backup method.

b. Safety and Relief Valve Bases:

Affected page: B 3/4.6-3

Description of the current requirements (**Safety and Relief Valve Bases**)

The pressure relief function of the Relief and Safety Valves, Bases 3/4.6.E/F (Page B 3/4.6-3) have been established to limit reactor vessel pressure to less than 110% of vessel design pressure. Evaluations have determined that the most severe transient is the closure of all the main steam isolation valves (MSIVs) followed by a reactor scram on high neutron flux.

Basis for the current requirements (**Safety and Relief Valve Bases**)

The ASME Boiler and Pressure Vessel Code requires the reactor pressure vessel to be protected from overpressure during upset conditions by self-actuated safety valves. The overpressure protection system must accommodate the most severe pressurization transient. The analysis results must demonstrate that the design safety valve capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of the reactor pressure vessel design pressure.

Need for revision of the requirements (**Safety and Relief Valve Bases**)

The change is proposed to reflect the possibility that the limiting pressurization event could result in a peak pressure higher than the ASME compliance event. The need to assure that peak pressure is below 110% of design pressure on a cycle specific basis is therefore stated for either type of transient.

Description of the revised requirements (**Safety and Relief Valve Bases**)

The wording in Bases 3/4.6.F is modified to read:

The overpressure protection system must accommodate the peak transient pressure during the most severe licensing basis pressurization transient. This includes, but is not limited to, the licensing basis ASME Section III compliance event which is the closure of all MSIVs with no credit for relief function or direct scram from valve position. For the purpose of the ASME Section III analysis, the SRV (combination safety/relief valve) is assumed to operate only in the safety mode. The ASME Section III analysis demonstrates that the design capacity of the SVs and SRV is capable of maintaining the reactor pressure below the ASME code limit. The licensing basis pressurization transients are evaluated for each reload to assure compliance with the ASME Code limit of 110% of vessel design pressure. This LCO ensures the acceptance limit of 1375 psig is met during the most severe licensing basis pressurization transient.

Basis for the revised requirements (**Safety and Relief Valve Bases**)

The Basis for Specifications 3/4.6.E and F is modified to reflect that, although unlikely, another pressurization event could produce higher peak pressure than the licensing basis event used to demonstrate compliance with ASME Section III. The licensing basis pressurization transients are analyzed on a cycle specific basis and compared with the ASME acceptance criteria for upset conditions.

c. References:

Affected page: 6-16

Description of the current requirements(References)

The Administrative Controls section of the Technical Specifications lists the NRC approved topicals for the analytical methods used to determine the operating limits (Specification 6.9.A.6) . The existing items listed are the General Electric Standard Application for Reactor Fuel (GESTAR), and ComEd Reports benchmarking BWR Nuclear Design methods and Neutronic Licensing Analyses.

Basis for the current requirements(References)

The list of documents provides documentation in the Specifications of the NRC approved methods used to determine operating limits. The details of the limits are provided in the Core Operating Limits Report (COLR). This list was created in response to Generic Letter 88-16.

Need for revision of the requirements(References)

The change from General Electric (GE) to Siemens Power Corporation (SPC) results in the need to include NRC approved SPC methods in the subject Reference list. Also included is ComEd Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods."

Description of the revised requirements(References)

The SPC Reload Licensing Methodology Summary document, EMF-94-217 (Attachment E), is the SPC equivalent of the GESTAR. However, it is not an NRC approved document. Therefore, a review of EMF-94-217 was performed and the following list of documents is proposed for inclusion in Specification 6.9.A.6.b.:

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

(2) Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, XN-NF-85-67(P)(A).

(3) Qualification of Exxon Nuclear Fuel for Extended Burnup: Extended Burnup Qualification of ENC 9x9 BWR Fuel, XN-NF-82-06(P)(A).

(4) Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advance Nuclear Fuels Corporation 9x9-IX and 9x9-9X BWR Reload Fuel, ANF-89-014(P)(A).

(5) Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A).

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

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

(8) Advanced Nuclear Fuels Critical Power Methodology for Boiling Water Reactors, ANF-524(P)(A).

(9) COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analysis, ANF-913(P)(A) Volume 1.

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

Basis for the revised requirements(References)

The revised list of documents provide reference to the NRC approved methodologies being used to determine the operating limits. Reference: NRC Generic Letter 88-16.

d. Editorial clarifications:

Affected pages: B 2-1, B 2-2, B 2-5, B 2-8, B 3/4.2-1, B 3/4.3-6, B 3/4.3-7, 5-5, 5-6

1 - water level and top of active fuel (pp. B 2-8, B 3/4.2-1)

2 - MCPR Bases (pp. B 2-1, B 2-2)

3 - Low Power RWE peak power level (p. B 2-5)

4 - Fuel Description (p. 5-5)

5 - Reactor Coolant System (pp. XIV, 5-6)

6 - Rod Worth Minimizer (pp. B 3/4.3-6, B 3/4.3-7)

Description of the current requirements (editorial)

1 - Water level and top of active fuel: Pages B 2-8 and B 3/4.2-1 of the Instrumentation Bases discusses the top of active fuel.

2 - MCPR Bases: Pages B 2-1 and B 2-2 discuss fuel damage calculations performed as part of the analyses of Anticipated Operational Occurrences (AOOs). The MCPR Safety Limit is presented as the protection against transition boiling.

3 - Low Power RWE peak power level : Page B 2-5 refers to the power level at which the low power RWE is terminated by the Intermediate Range Monitor (IRM) system. This Bases description relates the core average power level during the RWE event. It correctly reflects the 1% power level indication from UFSAR Figure 7.6-6, for the CORE AVERAGE power level. However, the LOCAL power level is more relevant to the RWE results.

4 - Fuel description: Page 5-5 describes the fuel used in the reactor.

5 - Reactor Coolant System (p. 5-6): Specification 5.4 (page 5-6) identifies key reactor coolant system design features. 4 - Fuel description: Page 5-5 describes the fuel used in the reactor.

6 - Rod Worth Minimizer: The Bases discussion (pages B 3/4.3-6 and B 3/4.3-7) for the Rod Worth Minimizer was clarified to state that the control rod sequences used during the cycle are not all written prior to cycle startup, but are verified to meet the 280 cal/gm limit up to the Low Power Set Point (LPSP) using NRC approved methodology. This methodology is referenced in Technical Specification 6.9.A.6, but is not described in detail

Basis for the current requirements (editorial)

1 - Water level and top of active fuel (pp. B 2-8, B 3/4.2-1): The water level limitations are to provide adequate core cooling by ensuring the core remains covered by water.

2 - MCPR Bases (pp. B 2-1, B 2-2): Calculations are performed to ensure the fuel cladding integrity limits are not exceeded during AOOs. Part of the analyses performed is to determine the MCPR Safety Limit.

3 - Low Power RWE wording change (p. B 2-5): The power level at which the IRMs would terminate a low power Control Rod Withdrawal Error (RWE) event is stated to be 1% of rated power. This presentation of power level is illustrative, and reflects the results of the UFSAR analysis of the results of the event. The avoidance of a MCPR Safety Limit violation is the basis for the IRM protection during this event.

4 - Fuel description (p. 5-5): The fuel assemblies are described in Specification 5.3.A. This description includes cladding and fuel material and discussion of NRC approved configurations and designs, including the use of lead test assemblies.

5 - Reactor Coolant System (p. 5-6): The design features of the reactor coolant system are provided to ensure appropriate review is performed when changing one of these parameters.

6 - Rod Worth Minimizer (pp. B 3/4.3-6, B 3/4.3-7): The RWM enforces the control rod sequence up to the LPSP to ensure the energy deposition from a control rod drop accident results in a peak fuel enthalpy less than 280 cal/gm.

Need for revision of the requirements (editorial)

1 - Water level and top of active fuel (pp. B 2-8, B 3/4.2-1): Clarification is added to page B 3/4.2-1 reflecting the slight variation in top of active fuel heights depending upon fuel design.

2 - MCPR Bases (pp. B 2-1, B 2-2): The wording is clarified on pages B 2-1 and B 2-2. The changes are editorial in nature.

3 - Low Power RWE wording change (p. B 2-5): The unblocked control rod withdrawal error (RWE) event is discussed in section 7.6.1 of the UFSAR. The power level cited at which the IRMs terminate the transient is 7.7% of rated power. The Bases on page B 2-5 cite a power level of 1%.

4 - Fuel description (p. 5-5): The SPC fuel design (ATRIUM-9B) contains a square water box. The wording in Specification 5.3.A should be modified to reflect this.

5 - Reactor Coolant System (p. 5-6): Modifications to the reactor coolant system, such as a physical modification to the coolant boundary or the introduction of new fuel types, are reviewed in accordance with 10CFR50.59 prior to implementation. UFSAR section 5 establishes the design requirements of the reactor coolant system. The parameters in Specification 5.4 are more appropriately located in a list of vessel parameters in UFSAR section 5, consistent with NUREG-1434.

6 - Rod Worth Minimizer (pp. B 3/4.3-6, B 3/4.3-7): The Bases discussion was clarified to state that the control rod sequences used during the cycle are not all written prior to cycle startup, but are verified to meet the 280 cal/gm limit up to the Low Power Set Point (LPSP) using NRC approved methodology. This methodology is referenced in Technical Specification 6.9.A.6, but is not described in detail. A clarifying wording change from "be worth enough" to "sufficient reactivity worth" was also made.

Description of the revised requirements (editorial)

1 - water level and top of active fuel (pp. B 2-8, B 3/4.2-1):

The Bases on page B 3/4.2-1 are modified to include the following:

> Current fuel designs incorporate slight variations in the length of the active fuel, and thus the actual top of active fuel, when compared with the original fuel designs. Safety Limits, instrument water level setpoints, and associated LCOs refer to the top of active fuel. In these cases, the top of active fuel is defined as 360 inches above vessel zero. Licensing analyses, both accident and transient, utilize this definition for the automatic initiation and manual intervention associated with these events.

2 - MCPR Bases (pp. B 2-1, B 2-2):

> The statement: "The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an AOO" is changed to read, "The fuel cladding integrity limit is set such that no fuel damage is calculated to occur as a result of an AOO."
>
> The statement: "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" is modified. It is clarified to read, "Therefore, the fuel cladding integrity Safety Limit is defined such that, with the limiting fuel assembly operating at the MCPR Safety Limit, more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. This includes consideration of the power distribution within the core and all uncertainties."

The statement on page B 2-2 that reads, "Because the transition boiling correlation is based on a significant quantity of practical test data, there is a very high confidence that operation of a fuel assembly at the condition where MCPR is equal to the fuel cladding integrity Safety Limit would not produce transition boiling" is clarified. It is changed to read, "Because the transition boiling correlation is based on a significant quantity of practical test data, there is a very high confidence that operation of a fuel assembly at the condition where MCPR is equal to the fuel cladding integrity Safety Limit would not produce fuel cladding failure due to overheating."

3 - Low Power RWE wording change (p. B 2-5):

The revised statement on page B 2-5 is changed to cite the LOCAL power level of 7.7% during the low power RWE event instead of the Core Average power level of 1% of rated power.

4 - Fuel Description (p. 5-5):

The second sentence of Specification 5.3.A. refers to the fuel containing water rods. It is modified to include reference to water boxes.

5 - Reactor Coolant System (p. 5-6):

The contents of Specification 5.4 (page 5-6) are proposed to be re located to the UFSAR. Page 5-6 is modified to read "[INTENTIONALLY BLANK]." This is consistent with page 5-7. The change is also reflected on Table of Contents page XIV.

6 - Rod Worth Minimizer (pp. B 3/4.3-6, B 3/4.3-7):

The Bases discussion was clarified to state that the control rod sequences used during the cycle are not all written prior to cycle startup, but are verified to meet the 280 cal/gm limit up to the Low Power Set Point (LPSP) using NRC approved methodology. This methodology is referenced in Technical Specification 6.9.A.6, but is not described in detail. A wording change from "be worth enough" to "have sufficient reactivity worth" was also made for clarification purposes.

Basis for the revised requirements (editorial)

1 - water level and top of active fuel (pp. B 2-8, B 3/4.2-1): Current fuel designs incorporate slight variations in the length of the active fuel, and thus the actual top of active fuel, when compared with the original fuel designs. Safety Limits, instrument water level setpoints, and associated LCOs refer to the top of active fuel. In these cases, the top of active fuel is defined as 360 inches above vessel zero. Licensing analyses, both accident and transient, utilize this definition for the automatic initiation and manual intervention associated with these events.

2 - MCPR Bases (pp. B 2-1, B 2-2):

The editorial changes clarify the Bases for the MCPR. The changes are editorial in nature.

3 - Low Power RWE wording change (p. B 2-5):

The existing Bases describes the power levels resulting from an unblocked control rod withdrawal error (RWE) in the low power range, with limiting instrumentation failure. The current description references the analysis results of 1% core average power as presented in UFSAR Figure 7.6-6. This figure and the associated UFSAR text also reference the local power level as reaching 7.7% when the event is terminated by the IRM initiated SCRAM. Since the Bases description is more relevant to the local power conditions than the core wide average, the appropriate local power level is implemented into the Technical Specification Bases. Because both of these numbers are in the current UFSAR, this change is only a clarification of the Bases.

4 - Fuel Description (p. 5-5):

The fuel description is modified to include reference to water boxes, which agrees more closely with the square water box in the new SPC fuel. This is compared with the existing cylindrical water rods in the current fuel design.

5 - Reactor Coolant System (p. 5-6):

This change is consistent with NUREG-1434. Configurations, design temperatures and pressures, and volumes of the reactor coolant system are detailed in the UFSAR. Section 5 of the UFSAR delineates the recirculation piping system design requirements. Any changes to these design parameters must conform to the requirements of 10CFR50.59. Therefore, re-locating these details from the Technical Specifications, while maintaining the details in the UFSAR, will not impact safe operation of the Facility.

6 - Rod Worth Minimizer (pp. B 3/4.3-6, B 3/4.3-7):

The three changes are editorial, consisting of a wording change, a clarification concerning the RDA methodology description referenced by Specification 6.9.A.6, and documents the actual development of control rod sequences throughout a cycle.

Schedule

This amendment is needed to support Quad Cities operations with Siemens fuel. Siemens fuel will be initially loaded into Quad Cities Unit 2 Cycle 15 (startup approximately April 1997) and Quad Cities Unit 1 Cycle 16 (startup approximately April 1998); therefore, this amendment is required prior to the startup of Quad Cities Unit 2 Cycle 15. It is requested that the amendment be approved by March 1, 1997, with the amendment to be implemented prior to the startup of Cycle 15 for Unit 2.

Attachment B

Summary of Proposed Changes