Florida Power & Light Company, 6351 S. Ocean Drive, Jensen Beach, FL 34957

FPL

July 19, 2000

L-2000-154
10 CFR 50.90

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

RE: St. Lucie Unit 1
Docket No. 50-335
Proposed License Amendment
Cycle 17 Reload

Pursuant to 10 CFR 50.90, Florida Power & Light Company (FPL) requests to amend Facility Operating License DPR-67 for St. Lucie Unit 1. The proposed amendment revises the St. Lucie Unit 1 license: i) to implement Siemens Power Corporation (SPC) high thermal performance (HTP) fuel assembly design in Cycle 17, ii) relocates shutdown margin (SDM) requirements in Modes 1 to 5 to the Core Operating Limits Report (COLR), iii) updates the COLR methodologies listed in the Technical Specification (TS) Section 6.9.1.11, and iv) requests relief from the SPC fuel assembly reconstitution restrictions for peripheral low power fuel assemblies. These changes involve modifying TS 3.1.1.1, 3.1.1.2, 3.1.2.2, 3.1.2.8, 6.9.1.11, and TS Bases for TS 2.1.1, 3/4.1.1.1, 3/4.1.1.2, 3/4.1.2, and 3/4.2.5. Applicable TS surveillance requirements are changed to be consistent with the proposed license amendment. Additionally, administrative changes are proposed to the boron concentration specifications related to the boration requirements in TS 3.1.1.1, 3.1.1.2, and 3.9.1.

This proposed license amendment adds Siemens Power Corporation Topical Report, EMF-1961(P), Revision 0, *Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors*, dated December 1998 to TS Section 6.9.1.11. This methodology, which is currently under NRC review, implements a statistical method for setpoint/transient analysis. It is proposed to be included in the TS if approved prior to NRC issuing this license amendment.

The proposed changes do not have any adverse impact on the plant safety or the operation of the plant at any power levels. The proposed changes do not impact the current cycle (Cycle 16) operation of St. Lucie Unit 1. The proposed methodology updates would be required to support the Cycle 17 reload analysis.

A001

Attachment 1 is a description of the changes and the safety analysis in support of the proposed amendment. Attachment 2 is the *Determination of No Significant Hazards Consideration*. Attachment 3 is a marked-up copy of the proposed Technical Specification changes.

The proposed amendment has been reviewed by the St. Lucie Facility Review Group and the Florida Power & Light Company Nuclear Review Board. In accordance with 10 CFR 50.91 (b)(1), a copy of the proposed amendment is being forwarded to the State Designee for the State of Florida.

Approval of this proposed license amendment is requested by January 31, 2001, to support restart of Unit 1 for Cycle 17, which is currently scheduled for April 2001. Please issue the amendment to be effective on date of issuance and to be implemented within 60 days of receipt by FPL.

Please contact us if there are any questions about this submittal.

Very truly yours,

Rajiv S. Kundalkar

Rajiv S. Kundalkar
Vice President
St. Lucie Plant

RSK/GRM

Attachments

cc: Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, St. Lucie Plant
Mr. William A. Passetti, Florida Department of Health and Rehabilitative Services

STATE OF FLORIDA)
) ss.
COUNTY OF ST. LUCIE)

Rajiv S. Kundalkar being first duly sworn, deposes and says:

That he is Vice President, St. Lucie Plant, for the Nuclear Division of Florida Power & Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information and belief, and that he is authorized to execute the document on behalf of said Licensee.

Rajiv S. Kundalkar

STATE OF FLORIDA

COUNTY OF ST. LUCIE

Sworn to and subscribed before me

this 19 day of JULY, 2000
by Rajiv S. Kundalkar, who is personally known to me.

Name of Notary Public - State of Florida

GEORGE R. MADDEN
Notary Public - State of Florida
My Commission Expires Jun 17, 2004
Commission # CC931387

(Print, type or stamp Commissioned Name of Notary Public)

ATTACHMENT 1

SAFETY ANALYSIS

Introduction

Florida Power and Light Company (FPL) proposes to amend the St. Lucie Unit 1 Operating License: i) to implement Siemens Power Corporation (SPC) high thermal performance (HTP) fuel assembly design in Cycle 17, ii) to relocate shutdown margin (SDM) requirements in Modes 1 to 5 to the Core Operating Limits Report (COLR), iii) to update the COLR methodologies listed in the Technical Specifications (TS), and iv) to obtain relief from the SPC fuel assembly reconstitution restrictions for peripheral low power fuel assemblies. These changes involve modifying TS 3.1.1.1, 3.1.1.2, 3.1.2.2, 3.1.2.8, 6.9.1.11, and TS Bases for TS 2.1.1, 3/4.1.1.1, 3/4.1.1.2, 3/4.1.2, and 3/4.2.5. Applicable TS surveillance requirements are changed to be consistent with the proposed license amendment. Additionally, administrative changes are proposed to the boron concentration specifications related to the boration requirements in TS 3.1.1.1, 3.1.1.2, and 3.9.1.

The evaluation of the proposed changes has demonstrated that the proposed changes would not have any adverse impact on the plant safety or on the operation of the plant at any power levels. The proposed changes do not impact the current cycle (Cycle 16) operation of St. Lucie Unit 1. The proposed methodology updates would be required to support Cycle 17 reload analysis.

Discussion of the Proposed Changes

2.1 Bases for TS 2.1.1 (REACTOR CORE)

The bases for the thermal margin limit lines Figure 2.1-1 are clarified with respect to the departure from nucleate boiling (DNB) correlation and the departure from nucleate boiling ratio (DNBR) limit applicable to the fuel assemblies in the core. The reference to "Table 2.1-1" is corrected to read "Table 2.2-1." Also the specification of DNBR limit of 1.22 is deleted.

2.2 TS 3.1.1.1 and 4.1.1.1.1: SHUTDOWN MARGIN – $T_{avg} > 200^0F$
Bases for TS 3/4.1.1.1 (SHUTDOWN MARGIN)

The shutdown margin limit in this specification is moved to the COLR. This TS is changed to delete reference to " $\geq$ 3600 pcm" and refer to the COLR for the shutdown margin limits. Also, the boration requirement of "1720 ppm" in the action statement is changed to "greater than or equal to 1720 ppm."

2.3 TS 3.1.1.2 and 4.1.1.2: SHUTDOWN MARGIN – $T_{avg} \leq 200\ ^0F$
Bases for TS 3/4.1.1.2 (SHUTDOWN MARGIN)

The shutdown margin limit in this specification is moved to the COLR. This TS is changed to delete reference to " ≥ 2000 pcm" and refer to the COLR for the shutdown margin limits. Also the boration requirement of "1720 ppm" in the action statement is changed to "greater than or equal to 1720 ppm."

2.4 TS 3.1.2.2 and 3.1.2.8: FLOW PATHS & BORATING WATER SOURCES
Bases for TS 3/4.1.2 (BORATION SYSTEMS)

The wording "at least 2000 pcm" in the action statement of TS 3.1.2.2 and 3.1.2.8 is changed to "the requirements of Specification 3.1.1.2."

Bases 3/4.1.2 is changed to delete "2000 pcm" and refer to the requirements of Specification 3.1.1.2.

2.5 TS 3.9.1: REFUELING OPERATIONS – BORON CONCENTRATION

The boration requirement of "1720 ppm" in the action statement is changed to "greater than or equal to 1720 ppm."

2.6 Bases for TS 3/4.2.5 (DNB PARAMETERS)

The specification of DNBR limit value of 1.22 is deleted.

2.7 TS 6.9.1.11: CORE OPERATING LIMITS REPORT (COLR)

Specifications 3.1.1.1 and 3.1.1.2 are added to TS 6.9.1.11.a. The methodology listed in TS 6.9.1.11.b is updated to include the following additional fuel vendor (SPC) methodologies approved by the NRC.

1. EMF-92-116(P)(A), Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Design," Siemens Power Corporation, February 1999.

2. EMF-92-153(P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation, March 1994.

3. EMF-96-029(P)(A) Volumes 1 and 2, "Reactor Analysis System for PWRs Volume 1 – Methodology Description, Volume 2 – Benchmarking Results," Siemens Power Corporation, January 1997.

The following methodology proposed to be included in TS 6.9.1.11 is currently under the NRC review.

4. EMF-1961(P), Revision 0, "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors," Siemens Power Corporation, December 1998.

2.8 FUEL ASSEMBLY RECONSTITUTION

The restrictions imposed on the SPC fuel assembly reconstitution methodology, documented in ANF-90-082(P)(A), for limiting the number of inert rods/guide tube per subchannel are proposed to be waived for core peripheral fuel assemblies. Additionally, it is proposed to allow solid stainless steel rods (no clad) as inert rods for reconstitution.

2.9 INDEX page xv and TS Section 6.9.2: SPECIAL REPORTS

Conforming changes to TS Index and to page 6-19c for movement of TS Section 6.92 to a new page 6-19c due to text addition to page 6-19b.

3.0 <u>Basis for Proposed Changes/Analysis of Impact on Safety</u>

3.1 Bases for TS 2.1.1 (REACTOR CORE)

The text in the bases for the thermal margin limit lines Figure 2.1-1 is modified to reflect the DNB correlation used in generating these limit lines. Since St. Lucie Unit 1 could have both the non-vaned and the HTP fuel designs in the core, conservative limit lines based on the XNB DNB correlation, as opposed to HTP DNB correlation, are maintained in Figure 2.1-1 as representative of the Reactor Core Safety Limit. Since the DNBR limit is inherently defined in the use of the correlation and this limit is different for the XNB and HTP DNB correlations, the value of this limit is deleted in the bases of TS 2.1.1. Use of the appropriate NRC approved DNB correlation is stipulated by TS 6.9.1.11. Cycle specific analysis will use the appropriate DNB correlation and the corresponding DNBR limit to ensure that the thermal margin DNBR limit is not violated for any anticipated combination of transient conditions initiated within the limiting conditions of operation in combination with the reactor protection systems.

"Table 2.1-1" is corrected to read "Table 2.2-1." This change is justified since Table 2.2-1 is the correct reference table for the specified trip setpoint.

3.2 TS 3.1.1.1 and 4.1.1.1.1: SHUTDOWN MARGIN – $T_{avg} > 200^0F$
TS 3.1.1.2 and 4.1.1.2: SHUTDOWN MARGIN – $T_{avg} \leq 200^0F$
Bases for TS 3/4.1.1.1 and 3/4.1.1.2 (SHUTDOWN MARGIN)

The proposed changes relocate the shutdown margin limits to the COLR. Moving the shutdown margin limits to the COLR provides the flexibility to optimize the SDM requirements based on cycle specific fuel management and design considerations, such as scram worth, burnable absorber loadings, soluble boron level, etc. This change is consistent with the St. Lucie Unit 2 specifications. There is no change to limits of the shutdown margin requirements due to this proposed amendment.

The shutdown margin requirement for T_{avg} greater than 200^0F (Modes 1 through 4) during times late-in-cycle is determined by the results of the Steam Line Break analysis. The specific late-in-cycle shutdown margin requirement may vary from cycle-to-cycle since the scram worth and the power distribution may vary substantially from cycle-to-cycle. Therefore, the proposed change will allow flexibility to accommodate cycle specific time-in-life shutdown margin requirements under the provisions of 10 CFR 50.59, and thereby, obviate the need for license amendment. Specification 6.9.1.11.c requires core operating limits, such as shutdown margin, to meet all applicable limits of the safety analysis.

The shutdown margin requirement for T_{avg} less than or equal to 200^0F (Mode 5) may vary from cycle-to-cycle based on cycle specific fuel management and design considerations. The proposed change will allow flexibility to accommodate cycle-to-cycle variations in shutdown margin requirements necessary to meet the design basis under the provisions of 10 CFR 50.59, and thereby, obviate the need for license amendment.

The proposed change to specify boration requirements to read "greater than or equal to 1720 ppm" is appropriate and consistent with the borated water sources specified in TS 3.1.2.8.

The changes proposed to the Bases 3/4.1.1.1 and 3/4.1.1.2 provide consistency with the proposed changes to the respective specifications. The technical basis of the specifications remains unchanged.

3.3 TS 3.1.2.2 AND & 3.1.2.8: FLOW PATHS & BORATING WATER SOURCES
Bases for TS 3/4.1.2 (BORATION SYSTEMS)

The changes proposed to these specifications and the bases provide consistency with the proposed changes to relocate shutdown margin requirements to the COLR. The technical bases of the specifications and the "Action" statements remain unchanged.

3.4 TS 3.9.1: REFUELING OPERATIONS – BORON CONCENTRATION

The proposed change to specify boration requirements to read "greater than or equal to 1720 ppm" is appropriate and consistent with the borated water sources specified in TS 3.1.2.7.

3.5 Bases for TS 3/4.2.5 (DNB PARAMETERS)

The specification of DNBR limit value of 1.22 is deleted. Since the DNBR limit is inherently defined in the use of the correlation and this limit is different for the XNB and HTP DNB correlations, the value of this limit is deleted in the bases of TS 3/4.2.5.

3.6 TS 6.9.1.11: CORE OPERATING LIMITS REPORT (COLR)

TS 6.9.1.11.a lists the specifications whose limits are defined in the COLR. The addition of the Specifications 3.1.1.1 and 3.1.1.2 to this list is justified based on the changes proposed in this license amendment to relocate shutdown margin requirements in Modes 1 to 5 to the COLR.

The methodologies included in TS 6.9.1.11.b have been previously approved by the NRC for the appropriate applications. The addition of methodologies to the current list in this section of TS is described below:

1. EMF-92-116(P)(A), Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Design," Siemens Power Corporation, February 1999.

   This methodology has been previously approved by the NRC for fuel assembly mechanical design and will be used to justify the use of the HTP fuel design for Cycle 17.

2. EMF-92-153(P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation, March 1994.

   This methodology has been previously approved by the NRC for thermal hydraulic analysis of HTP fuel using HTP DNB correlation.

3. EMF-96-029(P)(A) Volumes 1 and 2, "Reactor Analysis System for PWRs Volume 1 – Methodology Description, Volume 2 – Benchmarking Results," Siemens Power Corporation, January 1997.

   This methodology provides improved reactor neutronic analysis and has been previously approved by the NRC for PWR applications.

The following methodology proposed to be included in TS 6.9.1.11 is currently under the NRC review.

4. EMF-1961(P), Revision 0, "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors," Siemens Power Corporation, December 1998.

   This methodology, which is currently under the NRC review, implements statistical method for setpoint/transient analysis, and is proposed to be included in TS if approved prior to the approval of this license amendment.

3.7 FUEL ASSEMBLY RECONSTITUTION

St. Lucie Unit 1 has experienced, in the past, limited fuel failures in the peripheral fuel assemblies in locations adjacent to the guide tubes. The peripheral fuel assemblies are typically low powered and are not limiting from safety considerations. The proposed relief from the fuel assembly reconstitution restrictions will allow replacement of the fuel rods with inert solid steel rods near the guide tube locations found to be most susceptible for fretting failures. This change, if found necessary for implementation, will reduce the risk of reactor coolant radionuclide activity increase and subsequent potential adverse impacts.

The SPC methodology for fuel assembly reconstitution is described in the report ANF-90-082(P)(A) Revision 1 and Supplement 1, "Application of ANF Design Methodology for Fuel Assembly Reconstitution." A justification of the use of a different type of inert rod as a replacement for fuel rods (solid stainless steel) and an increased number of inert rods per subchannel than is allowed by the NRC safety evaluation report is provided below.

<u>Solid Steel Inert Rod</u>

SPC's topical report on fuel assembly reconstitution allows the use of inert replacement rods for fuel rods. The report specifically allows the use of stainless steel or Zircaloy slugs in standard fuel rod cladding for the inert rod designs. In addition to these configurations, prior to the approval of the topical report, SPC has used solid stainless steel rods and solid Zircaloy rods with geometries machined to approximate the geometry of the fuel rods in reconstituted fuel assemblies and in shield assemblies.

SPC has performed an evaluation and concluded that these solid material rods are equivalent to the inert rods described in the topical report and are also acceptable replacements for fuel rods. This evaluation included the assembly-specific safety, mechanical, and neutronic evaluations described in the reference report.

The use of solid stainless steel rods is not significantly different than using Zircaloy clad inert rods with Zircaloy or stainless steel slugs. Slightly more material exists in the solid rod, however, this is easily accounted for in the reference report methodology.

The solid stainless steel rod is manufactured from common materials (such as 304 or 304L stainless steel) already used in SPC fuel assemblies. While the stainless steel rod provides a different material exposure to the coolant than the Zircaloy clad inert rod, the irradiation and oxide properties are fully acceptable (properties and performance information is available in industry literature). The differences in the material properties between the stainless steel and the Zircaloy cladding are easily accounted for in the topical report methodology.

Increased Number of Inert Rods per Subchannel

The NRC safety evaluation report for the SPC topical report contains the following restriction:

> *"However, a licensee fuel reconstitution in accordance with this document should conform to certain limitations as described in TER Section 4.3: (1) BWR reconstituted assemblies are limited to 9 rods per assembly, and (2) PWR reconstituted assemblies are limited to two inert rods or one inert rod and a guide tube per subchannel, with a total of no more than 26 percent inert replacement rods per PWR assembly. There is one exception to these limitations and that is for assemblies located on the outer edge of the core where the outer inert rods are used for shielding the pressure vessel from irradiation damage and preventing embrittlement. The maximum limit of 26 percent inert rods per assembly, however, remains applicable to these core edge assemblies. Any further exceptions from these limitations will require review on a case-by-case basis."*

The NRC justification for the exception to the limitation on the number of inert rods and/or guide tubes in a PWR assembly per subchannel is applicable to any assembly on the outer edge of the core whether the edge assembly is for the purpose of shielding the vessel or not. SPC interprets the TER to indicate that the reason for the limitation on the number of inert rods and/or guide tubes in a PWR assembly per subchannel is due to a concern about the applicability of the DNB correlation to configurations with more inert rods. SPC concludes that the exception is granted for assemblies on the core edge due to the low power of these assemblies, these assemblies are not limiting from a DNB perspective, and thus the applicability of the DNB correlation is not a concern. The assemblies on the core outer edge operate at low power regardless of whether they are being used as shielding assemblies. It is therefore concluded that the SER restriction on the number of inert rods and/or guide tubes per subchannel does not apply to assemblies on the core outer edge.

3.8 INDEX page xv and TS Section 6.9.2: SPECIAL REPORTS

Administrative conforming change to TS index and movement of TS Section 6.9.2 SPECIAL REPORTS to a new page 6-19c due to text additions to page 6-19b.

ATTACHMENT 2

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

The proposed amendment revises the St. Lucie Unit 1 license: i) to implement Siemens Power Corporation (SPC) high thermal performance (HTP) fuel assembly design in Cycle 17, ii) relocates shutdown margin (SDM) requirements in Modes 1 to 5 to the Core Operating Limits Report (COLR), iii) updates the COLR methodologies listed in the Technical Specification (TS) Section 6.9.1.11, and iv) requests relief from the SPC fuel assembly reconstitution restrictions for peripheral low power fuel assemblies. These changes involve modifying TS 3.1.1.1, 3.1.1.2, 3.1.2.2, 3.1.2.8, 6.9.1.11, and TS Bases for TS 2.1.1, 3/4.1.1.1, 3/4.1.1.2, 3/4.1.2, and 3/4.2.5. Applicable TS surveillance requirements are changed to be consistent with the proposed license amendment. Additionally, administrative changes are proposed to the boron concentration specifications related to the boration requirements in TS 3.1.1.1, 3.1.1.2, and 3.9.1.

The standards used to arrive at a determination that a request for amendment involves a no significant hazards consideration are included in the Commission's regulation, 10 CFR 50.92, which states that no significant hazards considerations are involved if the operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Each standard is discussed as follows:

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment would allow the implementation of HTP fuel design for Cycle 17. The design of this fuel will be evaluated to meet all the mechanical, neutronics and thermal-hydraulics requirements, and acceptance criteria based on the approved methodology. The relocation of shutdown margin to the COLR and other proposed changes have no adverse impact on the operation of the plant and have no relevance to the accident initiators. There are no changes to the plant configuration, and thus the frequency of occurrence of previously analyzed accidents is not affected by the proposed changes. The changes proposed to the fuel reconstitution methodology would not impact the design acceptance criteria for the reconstituted fuel assemblies.

The proposed change for the relocation of shutdown margin to the COLR has no impact on current safety analyses and their consequences. Changes to the COLR limits will be controlled per Generic Letter 88-16 under the provisions of 10 CFR 50.59 and the requirements of TS 6.9.1.11.c. The application of the added methodology, which includes the approved HTP DNB correlation, would remain consistent with the design basis requirements and would not involve a significant increase in the consequences of design basis accidents. Other proposed TS and TS bases changes do not affect safety analysis results. The changes proposed to the fuel reconstitution methodology would not impact the safety analysis consequences as the changes are related to the non-limiting rod locations.

Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Use of the modified specification would not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed amendment updates the list of approved methodology in TS 6.9.1.11, relocates shutdown margin requirements to the COLR and requests relief for fuel reconstitution requirements. None of these changes would create the possibility of a new kind of accident since the reload analysis with these changes would continue to meet all applicable design limits. There is no change to plant configuration, systems or components which would create new failure modes. The modes of operation of the plant would remain unchanged.

Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Use of the modified specification would not involve a significant reduction in a margin of safety.

The proposed changes have no significant adverse impact on the safety analysis. As such, these changes would continue to provide margin to the acceptance criteria for specified acceptable fuel design limits (SAFDL), 10 CFR 50.46(b) requirements, primary and secondary overpressurization, peak containment pressure, potential radioactive releases, and existing limiting conditions for operation. The future use of updated approved methodologies will follow all design basis requirements to ensure that a safety margin to the acceptance criteria would continue to remain available for full power operation of St. Lucie Unit 1.

Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

Based on the above, we have determined that the proposed amendment does not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the probability of a new or different kind of accident from any previously evaluated, or (3) involve a significant reduction in a margin of safety; and therefore does not involve a significant hazards consideration.

Environmental Consideration

The proposed license amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The proposed amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite, and no significant increase in individual or cumulative occupational radiation exposure. FPL has concluded that the proposed amendment involves no significant hazards consideration and meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), an environmental impact statement or environmental assessment need not be prepared in connection with issuance of the amendment.

ATTACHMENT 3

St. Lucie Unit 1Marked-up Technical Specification Pages

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INDEX

ADMINISTRATIVE CONTROLS

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate below the level at which centerline fuel melting will occur. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related ~~to DNB through the Siemens~~ Power Corporation (SPC) XNB correlation. The XNB DNB correlation ~~has been developed~~ to predict the DNB flux and the location of DNB for axially uniform and non-~~uniform heat~~ flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

INSERT (A)

The minimum value of the DNBR during steady state operation, normal ~~operational transients,~~ and anticipated transients is limited to 1.22 using the XNB DNBR correlation. This ~~value corresponds~~ to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

INSERT (B)

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor ~~Coolant System~~ pressure and maximum cold leg temperature with four Reactor Coolant Pumps ~~operating for~~ which the minimum DNBR is no less than the DNBR limit for the axial shape shown in Figure B 2.1-1. ~~The limits in~~ Figure 2.1-1 were calculated for reactor coolant inlet temperatures less than or equal to 580°F. The dashed line at 580°F coolant inlet temperature is not a safety limit; however, operation above 580°F is not possible because of the actuation of the main steam line safety valves which limit the maximum value of reactor inlet temperature. Reactor operation at THERMAL POWER levels higher than 112% of RATED THERMAL POWER is prohibited by the high power level trip setpoint specified in Table ~~2.1-1~~ 2-2-1. The area of safe operation is below and to the left of these lines.

INSERT (C)

SAFETY LIMITS

BASES

The conditions ~~for the Thermal Margin Safety Limit curves in Figure 2.1-1~~ to ~~be valid~~ are shown on the figure.

The reactor protective system in combination with the Limiting Conditions for Operation is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure, and thermal power level that would result in a DNBR of less than the DNBR limit and preclude the existence of flow instabilities. INSERT D

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Reactor Coolant System piping, valves and fittings are designed to ANSI B 31.7, Class I which permits a maximum transient pressure of 110% (2750 psia) of component design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

INSERT A

DNB is not a directly measured parameter during operation and therefore THERMAL POWER, Reactor Coolant Temperature and Pressure have been related to DNB using a DNB correlation developed to predict the Critical Heat Flux (CHF) for DNB. The CHF is the heat flux at a particular core location that would cause DNB. The ratio of the CHF to the actual local heat flux at a particular core location is called the DNB Ratio (DNBR) and is indicative of the margin to DNB.

INSERT B

The minimum allowed value of the DNBR during steady state operation, normal operational transients, and anticipated transients is the DNBR limit from the appropriate DNB correlation. The DNBR limit corresponds to a 95% probability at a 95% confidence level that DNB will not occur at a particular core location, providing appropriate margin to DNB for all operating conditions. In a core with fuel assemblies of different designs (mixed core), there may be more than one DNB correlation and associated DNBR limit that defines DNB for the core.

INSERT C

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure, and maximum cold leg temperature with four Reactor Coolant Pumps operating for which the DNBR limit corresponding to the Siemens Power Corporation (SPC) XNB DNB correlation is not violated for the following conditions:

1. reactor coolant inlet temperatures less than or equal to 580^0F,
2. THERMAL POWER less than or equal to 112%,
3. reactor coolant vessel flow of 365,000 gpm, and
4. the axial power shape shown on Figure B2.1-1.

INSERT D

Specific verification of the DNBR limit with an appropriate DNB correlation ensures that the Reactor Core Safety Limit is satisfied.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} ≥ 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be ~~≥ 3600 pcm~~ Insert within the limits specified in the COLR.

APPLICABILITY: MODES 1, 2*, 3 and 4.

ACTION:

not within limits

With the SHUTDOWN MARGIN ~~< 3600 pcm~~ immediately initiate and continue boration at ≥ 40 gpm of 1720 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

Insert greater than or equal to

within the COLR limits

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be ~~≥ 3600 pcm~~:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is not fully inserted, and is immovable as a result of excessive friction or mechanical interference or is known to be untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. When in MODES 1 or 2#, at least once per 12 hours by verifying that CEA group withdrawal is within the Power Dependent Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2## at least once during CEA withdrawal and at least once per hour thereafter until the reactor is critical.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the CEA groups at the Power Dependent Insertion Limits of Specification 3.1.3.6.

* See Special Test Exception 3.10.1.
\# With Keff ≥ 1.0.
\## With Keff < 1.0.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be: within the limits specified in the COLR ~~≥ 2000 pcm~~, and in addition with the Reactor Coolant System drained below the hot leg centerline, one charging pump shall be rendered inoperable.*

APPLICABILITY: MODE 5.

ACTION:

If the SHUTDOWN MARGIN requirements cannot be met, immediately initiate and continue boration at ≥ 40 gpm of greater than or equal to 1720 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

Insert

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN requirements of Specification 3.1.1.2 shall be determined:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. At least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

c. At least once per 24 hours, when the Reactor Coolant System is drained below the hot leg centerline, by consideration of the factors in 4.1.1.2.b and by verifying at least one charging pump is rendered inoperable.*

* Breaker racked-out.

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. One flow path from the boric acid makeup tank(s) with the tank meeting Specification 3.1.2.8 part a) or b), via a boric acid makeup pump through a charging pump to the Reactor Coolant System.

b. One flow path from the boric acid makeup tank(s) with the tank meeting Specification 3.1.2.8 part a) or b), via a gravity feed valve through a charging pump to the Reactor Coolant System.

c. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System.

OR

At least two of the following three boron injection flow paths shall be OPERABLE:

a. One flow path from each boric acid makeup tank with the combined tank contents meeting Specification 3.1.2.8 c), via both boric acid makeup pumps through a charging pump to the Reactor Coolant System.

b. One flow path from each boric acid makeup tank with the combined tank contents meeting Specification 3.1.2.8 c), via both gravity feed valves through a charging pump to the Reactor Coolant System.

c. The flow path from the refueling water storage tank, via a charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or make the reactor subcritical within the next 2 hours and borate to a SHUTDOWN MARGIN equivalent to ~~at least 2000 pcm~~ at 200°F; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

the requirements of Specification 3.1.1.2

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 At least two of the following four borated water sources shall be OPERABLE:

a. Boric Acid Makeup Tank 1A in accordance with Figure 3.1-1.

b. Boric Acid Makeup Tank 1B in accordance with Figure 3.1-1.

c. Boric Acid Makeup Tanks 1A and 1B with a minimum combined contained borated water volume in accordance with Figure 3.1-1.

d. The refueling water tank with:

   1. A minimum contained volume of 401,800 gallons of water,
   2. A minimum boron concentration of 1720 ppm,
   3. A maximum solution temperature of 100°F,
   4. A minimum solution temperature of 55°F when in MODES 1 and 2, and
   5. A minimum solution temperature of 40°F when in MODES 3 and 4.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one borated water source OPERABLE, restore at least two borated water sources to OPERABLE status within 72 hours or make the reactor subcritical within the next 2 hours and borate to a SHUTDOWN MARGIN equivalent to ~~at least 2000 ppm~~ the requirements of Specification 3.1.1.2 at 200°F; restore at least two borated water sources to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 At least two borated water sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

   1. Verifying the boron concentration in each water source,

3/4.9 REFUELING OPERATIONS

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head unbolted or removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling cavity shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at $\geq$ 40 gpm of 1720 ppm boron or its equivalent to restore boron concentration to within limits.

Insert (greater than or equal to)

SURVEILLANCE REQUIREMENTS

4.9.1.1 The boron concentration limit shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length CEA in excess of 3 feet from its fully inserted position.

4.9.1.2 The boron concentration of the refueling cavity shall be determined by chemical analysis at least 3 times per 7 days with a maximum time interval between samples of 72 hours.

* The reactor shall be maintained in MODE 6 when the reactor vessel head is unbolted or removed.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

(as specified in the COLR for Specification 3.1.1.1)

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN ~~of 5000 pcm~~ is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN required by Specification 3.1.1.1 is based upon this limiting condition and is consistent with FSAR accident analysis assumptions. For earlier periods during the fuel cycle, this value is conservative. With $T_{avg} \leq 200°F$, the reactivity transient resulting from a boron dilution event with a partially drained Reactor Coolant System requires a ~~2000 pcm~~ SHUTDOWN MARGIN, and restrictions on charging pump operation to provide adequate protection. ~~A 2000 pcm~~ This SHUTDOWN MARGIN is 1000 pcm conservative for Mode 5 operation with total RCS volume present, however LCO 3.1.1.2 is written conservatively for simplicity.

Insert (as specified in the COLR for Specification 3.1.1.2)

3/4.1.1.3 BORON DILUTION AND ADDITION

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration changes in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 11,400 cubic feet in approximately 26 minutes. The reactivity change rate associated with boron concentration changes will be within the capability for operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limiting values of the MTC ensure that the assumptions for the MTC used in the accident and transient analyses remain valid through each fuel cycle. Determination of MTC at the specified conditions ensures that the maximum positive and/or negative values of the MTC will not exceed the limiting values.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

The MTC is expected to be slightly negative at operating conditions. However, at the beginning of the fuel cycle, the MTC may be slightly positive at operating conditions and since it will become more positive at lower temperatures, this specification is provided to restrict reactor operation when T_{avg} is significantly below the normal operating temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid pumps, and 5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions ~~of 2000 pcm~~ after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions. This requirement can be met for a range of boric acid concentrations in the Boric Acid Makeup Tanks (BAMTs) and Refueling Water Tank (RWT). This range is bounded by 5400 gallons of 3.5 weight percent (6119 ppm boron) boric acid from the BAMTs and 17,000 gallons of 1720 ppm borated water from the RWT to 8700 gallons of 2.5 weight percent (4371 ppm boron) boric acid from the BAMTs and 13,000 gallons of 1720 ppm borated water from the RWT. A minimum of 45,000 gallons of 1720 ppm boron is required from the RWT if it is to be used to borate the RCS alone.

The requirements for a minimum contained volume of 401,800 gallons of borated water in the refueling water tank ensures the capability for borating the RCS to the desired level. The specified quantity of borated water is consistent with the ECCS requirements of Specification 3.5.4. Therefore, the larger volume of borated water is specified here too.

Corresponding to the requirements of Specification 3.1.1.2

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

POWER DISTRIBUTION LIMITS

BASES

the DNB Margin LCO, and Thermal Margin/Low Pressure LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. If F_r^T or T_q exceed their basic limitations, operation may continue under the additional restrictions imposed by the ACTION statements since these additional restrictions provide adequate provisions to assure that the assumptions used in establishing the Linear Heat Rate, Thermal Margin/Low Pressure and Local Power Density - High LCOs and LSSS setpoints remain valid. An AZIMUTHAL POWER TILT > 0.10 is not expected and if it should occur, subsequent operation would be restricted to only those operations required to identify the cause of this unexpected tilt.

The requirement that the measured value of $(1+T_q)$ be multiplied by the calculated value of F_r to determine F_r^T is applicable only when F_r is calculated with a non-full core power distribution analysis. With a full core power distribution analysis code the azimuthal tilt is explicitly accounted for as part of the radial power distribution used to calculate F_r.

The surveillance requirements for verifying that F_r^T and T_q are within their limits provide assurance that the actual values of F_r^T and T_q do not exceed the assumed values. Verifying F_r^T after each fuel loading prior to exceeding 75% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the safety analyses assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR ~~of ≥ 1.22~~ greater than or equal to the DNBR limit throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

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ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (continued)

6.9.1.9 At least once every 5 years, an estimate of the actual population within 10 miles of the plant shall be prepared and submitted to the NRC.

6.9.1.10 At least once every 10 years, an estimate of the actual population within 50 miles of the plant shall be prepared and submitted to the NRC.

6.9.1.11 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

INSERT (E)

Specification 3.1.1.4	Moderator Temperature Coefficient
Specification 3.1.3.1	Full Length CEA Position – Misalignment > 15 inches
Specification 3.1.3.6	Regulating CEA Insertion Limits
Specification 3.2.1	Linear Heat Rate
Specification 3.2.3	Total Integrated Radial Peaking Factor – F_r^T
Specification 3.2.5	DNB Parameters
Specification 3.9.1	Refueling Operations – Boron Concentration

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, as described in the following documents or any approved Revisions and Supplements thereto:

1. WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988 (Westinghouse Proprietary)

2. NF-TR-95-01, "Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants," Florida Power & Light Company, January 1995.

3. XN-75-27(A) and Supplements 1 through 5, [also issued as XN-NF-75-27(A)], "Exxon Nuclear Neutronic(s) Design Methods for Pressurized Water Reactors," Exxon Nuclear Company, Inc. / Advanced Nuclear Fuels Corporation, Report and Supplement 1 dated April 1977, Supplement 2 dated December 1980, Supplement 3 dated September 1981 (P), Supplement 4 dated December 1986 (P), and Supplement 5 dated February 1987 (P)

4. ANF-84-73(P)(A) Revision 5, Appendix B, & Supplements 1 and 2, "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Advanced Nuclear Fuels Corporation, October 1990

5. XN-NF-82-21(P)(A) Revision 1, "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company, Inc., September 1983

6. a) ANF-84-93(P)(A) and Supplement 1, [also issued as XN-NF-84-93(P)(A)], "Steamline Break Methodology for PWRs," Advanced Nuclear Fuels Corporation, March 1989

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT (continued)

11. d) XN-NF-85-16(P)(A) Volume 1, and Supplements 1, 2 and 3; Volume 2, Revision 1 and Supplement 1, "PWR 17x17 Fuel Cooling Test Program," Advanced Nuclear Fuels Corporation, February 1990

e) XN-NF-85-105(P)(A) and Supplement 1, "Scaling of FCTF Based Reflood Heat Transfer Correlation for Other Bundle Designs," Advanced Nuclear Fuels Corporation, January 1990.

f) EMF-2087(P)(A) Revision 0, "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," Siemens Power Corporation, June 1999.

12. XN-NF-82-06(P)(A) Revision 1, and Supplements 2, 4 and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company, Inc., October 1986

13. ANF-88-133(P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation, December 1991

14. XN-NF-85-92 (P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company, Inc., November 1986

15. ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation, May 1992

16. XN-NF-507(P)(A), Supplements 1 and 2, "ENC Setpoint Methodology for C. E. Reactors: Statistical Setpoint Methodology," Exxon Nuclear Company, Inc., September 1986

INSERT F

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SHUTDOWN MARGIN, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC within the time period specified for each report.

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Specification 3.1.1.1	Shutdown Margin – T_{avg} Greater Than 200^0F
Specification 3.1.1.2	Shutdown Margin – T_{avg} Less Than or Equal to 200^0F

INSERT F

17. EMF-92-116(P)(A), Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Design," Siemens Power Corporation, February 1999.

18. EMF-92-153(P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation, March 1994.

19. EMF-96-029(P)(A) Volumes 1 and 2, "Reactor Analysis System for PWRs Volume 1 – Methodology Description, Volume 2 – Benchmarking Results," Siemens Power Corporation, January 1997.

20. EMF-1961(P), Revision 0, "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors," Siemens Power Corporation, December 1998.