Kewaunee Nuclear Power Plant
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NMC
Committed to Nuclear Excellence

Kewaunee / Point Beach Nuclear
Operated by Nuclear Management Company, LLC

NRC-02-108

December 19, 2002

10 CFR 50.90

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

Ladies/Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
Response to Request for Additional Information Related to Proposed Revision to the Kewaunee Nuclear Power Plant Technical Specifications LAR 185, "Core Operating Limits Report"

References:

1) Letter from Mark E. Warner (NMC) to Document Control Deck (NRC), "License Amendment Request 185 to the Kewaunee Nuclear Power Plant Technical Specifications, 'Core Operating Limits Report'", dated July 26, 2002.

2) Letter from John G. Lamb (NRC) to Mark E. Warner (NMC), "Kewaunee Nuclear Power Plant – Request for Additional Information Related to "License Amendment Request 185 to the Kewaunee Nuclear Power Plant Technical Specifications, 'Core Operating Limits Report'", dated November 21, 2002.

In reference 2, the Nuclear Regulatory Commission (NRC) staff requested additional information, within 30 days, concerning license amendment request (LAR) 185 to the Kewaunee Nuclear Power Plant Technical Specifications, 'Core Operating Limits Report'", (Reference 1). This letter is the Nuclear Management Company, LLC. (NMC) response to the NRC's request for additional information. The purpose of this letter is to respond to the NRC staffs request for additional information and to revised selected TS and COLR pages for replacement of these pages in NMC's reference 1 submittal.

Attachment 1 to this letter contains the questions the NRC staff requested with NMC's responses.
Attachment 2 contains the revised TS strike-out pages.
Attachment 3 contains the revised TS pages.
Attachment 4 contains the revised core operating limits report (COLR) pages.
Attachment 5 contains the revised list of methods used for the development of the COLR parameters.

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As the changes do not alter the conclusions reached in NMC's reference 1 submittal, the safety analysis, significant hazards determination, and the environmental considerations statements contained in reference 1 are still applicable and support the changes contained herein.

NMC requests approval of this license amendment request in accordance with the date contained in reference 1. This submittal contains no commitments. If your have any questions concerning this submittal please contact Mr. Gerald Riste at (920)388-8424.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on December 19, 2002.

Sincerely,

Thomas Coutu

Thomas Coutu
Site Vice President
Kewaunee Nuclear Power Plant

GOR

Attachments

1. NMC response to NRC request for additional information
2. KNPP TS Strike-out pages
3. KNPP TS affected pages
4. KNPP affected COLR page
5. KNPP Methods Matrix

cc - US NRC, Region III
US NRC Senior Resident Inspector
Electric Division, PSCW

ATTACHMENT 1

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Dated

December 19, 2002

NMC Responses
To
NRC Request for Additional Information

NRC Question #1

Your proposed TS 1.0.r.b, "Shutdown Margin (SDM)," states that the SDM is calculated assuming the fuel and moderator temperatures are changed to the program temperature. NUREG-1431, Vol. 1, Rev. 2, "Standard Technical Specifications, Westinghouse Plants," on the other hand, states that the SDM should be calculated assuming the temperatures are changed to the "nominal zero power design level." What is the difference between "program temperature" and "nominal zero power design level?" If there is a difference, why is it acceptable to use "program temperature" for establishing SDM?

NMC Response

The "nominal zero power design level" and the "program temperature" were considered to be the same. The term level was changed to temperature to ensure there was no confusion as to which design level was identified. To avoid confusion between the NRC staff and NMC, the statement will be change to say "nominal zero power design temperature."

Additionally, NMC reviewed the outline hierarchy of TS 1.0.r and found it different from the normal convention used in the rest of KNPP TS. To correct this inconsistency the two items under TS 1.0.r are changed to TS 1.0.r.1 and TS 1.0.r.2. These changes are administrative in nature and preset no safety concerns.

NRC Question #2

In TS 2.1, "Safety Limits, Reactor Core," you added peak fuel centerline limits. NUREG-1431 and WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report," both suggest using a peak fuel centerline temperature adjusted for fuel burnup. How did you determine your proposed limit of 4700°F and why is this value acceptable? Also, up to what levels of fuel burnup did you account for when determining this value?

NMC Response

NUREG 1431, "Standard Technical Specifications Westinghouse Plants," Technical Specification 2.1.1.2 states:

> The peak fuel centerline temperature shall be maintained < [5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup].

Kewaunee Nuclear Power Plant Operating License, DPR-43, states, "The maximum rod average burnup for any rod shall be limited to 60 GWD/MTU until completion of an NRC environmental assessment supporting an increased limit."[1] Subsequent to the completion of the NRC environmental assessment, NMC may submit a license amendment request to increase the maximum rod average burnup limit to 62 GWD/MTU (62,000 MWD/MTU).

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[1] KNPP Operating License, DPR-43, item 2.C(5), Revision 9/24/02, Amendment 163

If a margin of 3,000 MWD/MTU is added to the maximum burnup limit of 62 GWD/MTU and this is inserted in the aforementioned formula this results in peak centerline temperature of approximately 4700°F (5080°F – 58°F(65000/10000) = 4703°F). Thus, the limit of 4700°F was placed in KNPP TS.

NRC Question #3

WCAP-14483-A requires that the COLR analytical methods be provided in your TSs. What analytical method did you use to calculate the OTDT and OPDT trip setpoints? Is this method included in your TSs? Are there any required analytical methods missing from your proposed TSs?

NMC Response

To determine the analytical methods used to calculate the OTDT and OPDT trip setpoints NMC searched back to the KNPP Technical Specifications in affect in 1974. This review showed that the equations and the constants used in the equation and method of calculating the trip setpoints for OTDT and OPDT have not changed since 1974. As such the trip setpoints equations, and constants associated with OTDT and OPDT have been approved by the NRC since initial licensing.

During initial licensing the method used for determining these setpoints was consistent with the method contained in WCAP-8745-P-A, "Design Basis For The Thermal Overpower Delta-T And Thermal Overtemperature Delta-T Trip Functions," September 1986. WCAP-8745-P-A was written in 1977, after the KNPP setpoints were determined, and approved by the NRC in 1986. The WCAP that would be used for future changes the OTDT and OPDT trip setpoints would be WCAP-8745-P-A, "Design Basis For The Thermal Overpower Delta-T And Thermal Overtemperature Delta-T Trip Functions, September 1986". This method was omitted in NMC's original submittal because it has not been used. It has since been included in the revised TS pages, attached, delineating the method to be used.

From our review of GL 88-16 and WCAP 14483-A, there are no other required analytical methods missing from our proposed TSs. NMC conducted a review of the requirements for the methods to be contained in TS. Generic Letter (GL) 88-16, states that the methods to be used for determining the core operating limits shall be those previously reviewed and approved by the NRC. Those are listed in attachment 5. In WCAP 14483-A, "Generic Methodology for Expanded Core Operating Limits Report," WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," dated July 1985 is also listed and is used for the DNB Limit parameters. At KNPP, these parameters are controlled through NRC approved methods (WPSRSEM-NP-A, Revision 3) which is listed in the TS methods section. Attachment 5 is a table of the relocated parameters and the methods used to determine those parameters.

NRC Question #4

NUREG-1431 shows that the maximum upper limit for Moderator Temperature Coefficient (MTC) needs to be included in the TSs. Your proposed TS 3.1.f.3 does not include this limit. Why is this proposal acceptable?

NMC Response

The basis for changing the MTC values in KNPP TS and relocating these values to the COLR was (GL) 88-16, "Guidance for Technical Specification Changes for Cycle–Specific Parameter Limits". As MTC is a cycle–specific parameter that may change depending on the core reload physics, it was included in the relocation allowance of GL 88-16. TS 3.1.f.3 requires the MTC to be within specified limits of the COLR to ensure that the core operates within the assumptions of the accident analysis.

NUREG 1431 bases states that the purpose of placing the MTC value in TS is to establish a maximum positive value that cannot be exceeded. This MTC value is then not associated with the accident analysis but only sets upper bounds for the value, which NRC approval would be required to exceed. As the industry standard has an upper limit of MTC in TS, NMC will place a MTC upper limit in KNPP's TS. The KNPP TS will be worded similar to that found in standard technical specifications.

NRC Question #5

Your proposed Figure 2 of the COLR does not match the original Figure TS 3.10-1. Specifically, the required shutdown reactivity at 0 full power equilibrium boron concentration increased from 2000 pcm to 2200 pcm. Why is this change acceptable? Did you perform a 50.59 evaluation for this change?

NMC Response

This change from 2000 ppm to 2200 ppm is an administrative error. The limit of 2000 ppm is the correct value and the figure has been corrected in this submittal.

NRC Question #6

Kewaunee TS 3.8.a.5, "Refueling Operations," requires that a shutdown margin of greater than 5% Dk/k be maintained during refueling operations. However, you did not retain this requirement in your proposed COLR. Explain why this omitting this value from the COLR is acceptable.

NMC Response

NMC removed this requirement from TS 3.8.a.5 because it was considered redundant to the Refueling mode definition.

KNPP TS 1.0.j, "Modes," defines the Refueling mode as:

MODE	REACTIVITY Δk/k	COOLANT TEMP Tavg °F	FISSION POWER %
REFUELING	≤ –5%	≤ 140°F	~ 0%
COLD SHUTDOWN	≤ -1%	≤ 200	~0
INTERMEDIATE SHUTDOWN	(1)	> 200 < 540	~0
HOT SHUTDOWN	(1)	≥ 540	~0
HOT STANDBY	< 0.25%	~T_{oper}	< 2
OPERATING	< 0.25%	~T_{oper}	≥ 2
LOW POWER PHYSICS TESTING	(To be specified by specific tests)		
(1) Refer to Figure TS 3.10-1			

Current KNPP TS 3.8.a.5 states that during refueling operations:

> When there is fuel in the reactor, a minimum boron concentration of 2200 ppm and a shutdown margin of ≥ 5% Δk/k shall be maintained in the Reactor Coolant System during reactor vessel head removal or while loading and unloading fuel from the reactor. The required boron concentration shall be verified by chemical analysis daily.

Stating in TS 3.8.a.5 that during refueling operations a shutdown margin of ≥ 5% Δk/k shall be maintained in the Reactor Coolant System was considered redundant to the reactivity requirement of the mode definition of refueling. Because of this redundancy NMC removed the ≥ 5% Δk/k from TS 3.8.a.5.

As KNPP is already bound by this requirement in the refueling mode definition, NMC will place this requirement in the COLR to ensure the requirement is understood.

ATTACHMENT 2

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Dated

December 19, 2002

Strike-out TS Pages

TS 1.0-6

TS 3.1-10

TS 6.9-4

p. DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 is that concentration of I-131 (μ Ci/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be as listed and calculated with the methodology established in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

DOSE CONVERSION FACTOR	ISOTOPE
1.0000	I-131
0.0361	I-132
0.2703	I-133
0.0169	I-134
0.0838	I-135

q. CORE OPERATING LIMITS REPORT (COLR)

The COLR is the unit specific document that provides cycle-specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 6.9.a.4. Plant operation within these limits is addressed in individual Specifications.

r. SHUTDOWN MARGIN (SDM)

SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

1. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all RCCAs verified fully inserted by two independent means (TS 3.10.e), it is not necessary to account for a stuck RCCA in the SDM calculation. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM, and

2. In the OPERATING and HOT STANDBY MODES, the fuel and moderator temperatures are changed to the nominal zero power design temperature.

a. Minimum Conditions for Criticality

1. The reactor shall not be brought to a critical condition until the pressure-temperature state is to the right of the criticality limit line shown in Figure TS 3.1-1.

2. The reactor shall be maintained subcritical by at least 1% $\Delta k/k$ until normal water level is established in the pressurizer.

3. When the reactor is critical ~~and ≤ 60% RATED POWER,~~ the moderator temperature coefficient shall be ~~≤ 5.0 pcm/°F~~ as specified in the COLR, except during LOW POWER PHYSICS TESTING. ~~When the reactor is > 60% RATED POWER, the moderator temperature coefficient shall be zero or negative.~~ The maximum upper moderator temperature coefficient limit shall be ≤ 5 pcm/°F for power levels ≤ 60% RATED POWER and ≤ 0 pcm/°F for power levels > 60% RATED POWER.

~~4. The reactor will have a moderator temperature coefficient no less negative than -8 pcm/°F for 95% of the cycle time at full power.~~

4. If the limits of 3.1.f.3 cannot be met, then power operation may continue provided the following actions are taken:

   A. Within 24 hours, develop and maintain administrative control rod withdrawal limits sufficient to restore the moderator temperature coefficient to within the limits specified in TS 3.1.f.3. These withdrawal limits shall be in addition to the insertion limits specified in TS 3.10.d.

   B. If the actions specified in TS 3.1.f.~~5~~4.A are not satisfied, then be in HOT STANDBY within the next 6 hours.

~~(3) Nissley, M.E. et. al., "Westinghouse Large-Break LOCA Best-Estimate Methodology," WCAP-10924-P-A, Volume 1, Revision 1, Addendum 4, March 1991, Volume 1: Model Description and Validation; Addendum 4: Model Revisions.~~

~~(4) N. Lee et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081-NP-A (Non-Proprietary), dated August 1985.~~

(5) ~~C.M. Thompson, et al., "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condesation Model," WCAP-10054-P-A, Addendum 2, Revision 1 (Proprietary) and WCAP-10081-NP (Non-Proprietary), dated July 1997.~~

(6) ~~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, dated October 1986.~~

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

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

(9) ~~XN-NF-77-57, Exxon Nuclear Power Distribution Control for Pressurized Water Reactors, Phase II, dated January 1978, and Supplement 2, dated October 1981.~~

~~(10) WCAP-8745-P-A, "Design Basis for the Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Function", dated September 1986.~~

~~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 SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.~~

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

ATTACHMENT 3

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Dated

December 19, 2002

Affected TS Pages

TS 1.0-6

TS 3.1-10

TS 6.9-4

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DOSE EQUIVALENT I-131 is that concentration of I-131 (μ Ci/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be as listed and calculated with the methodology established in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

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SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

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3. When the reactor is critical the moderator temperature coefficient shall be as specified in the COLR, except during LOW POWER PHYSICS TESTING. The maximum upper moderator temperature coefficient limit shall be ≤ 5 pcm/°F for power levels ≤ 60% RATED POWER and ≤ 0 pcm/°F for power levels > 60% RATED POWER.

4. If the limits of 3.1.f.3 cannot be met, then power operation may continue provided the following actions are taken:

   A. Within 24 hours, develop and maintain administrative control rod withdrawal limits sufficient to restore the moderator temperature coefficient to within the limits specified in TS 3.1.f.3. These withdrawal limits shall be in addition to the insertion limits specified in TS 3.10.d.

   B. If the actions specified in TS 3.1.f.4.A are not satisfied, then be in HOT STANDBY within the next 6 hours.

(3) Nissley, M.E. et, al., "Westinghouse Large-Break LOCA Best-Estimate Methodology," WCAP-10924-P-A, Volume 1, Revision 1, Addendum 4, March 1991, Volume 1: Model Description and Validation; Addendum 4: Model Revisions.

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(5) C.M. Thompson, et al., "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condesation Model," WCAP-10054-P-A, Addendum 2, Revision 1 (Proprietary) and WCAP-10081-NP (Non-Proprietary), dated July 1997.

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

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

(9) XN-NF-77-57, Exxon Nuclear Power Distribution Control for Pressurized Water Reactors, Phase II, dated January 1978, and Supplement 2, dated October 1981.

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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 SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

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

ATTACHMENT 4

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Dated

December 19, 2002

Affected COLR Pages

Page 7 of 12

Figure 2

2.11 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

2.11.1 During steady state power operation, Tavg shall be < 568.8°F.

2.11.2 During steady state power operation, Pressurizer Pressure shall be ≥ 2205 psig

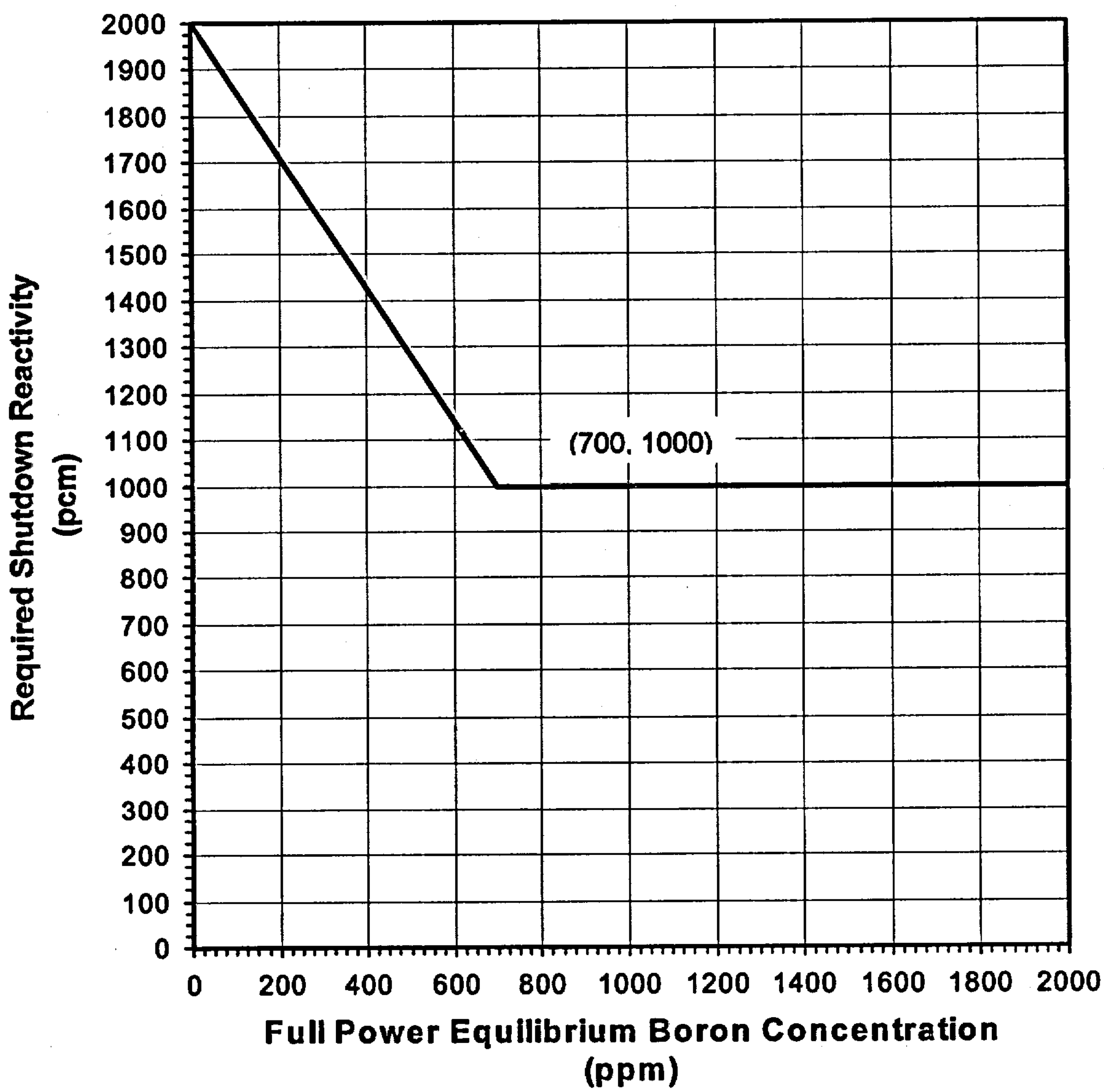
2.11.3 During steady state power operation, reactor coolant flow rate shall be ≥ 93,000 gpm per loop.

2.12 Refueling Boron Concentration

2.12.1 When there is fuel in the reactor, a minimum boron concentration of 2200 ppm and a shutdown margin of ≥ 5% Δ k/k shall be maintained in the Reactor Coolant System during reactor vessel head removal or while loading and unloading fuel from the reactor.

Figure 2

Required Shutdown Reactivity vs Boron Concentration



ATTACHMENT 5

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

Dated

December 19, 2002

KNPP Methods Matrix

Relocated Parameter and Approved Methods	
Parameter	NRC Approved Methodology
Reactor Core Safety Limits	SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION ON "QUALIFICATIONS OF REACTOR PHYSICS METHODS FOR APPLICATION TO KEWAUNEE" REPORT, dated August 21, 1979, report date September 29, 1978 KEWAUNEE NUCLEAR POWER PLANT – REVIEW FOR KEWAUNEE RELOAD SAFETY EVALUATION METHODS TOPICAL REPORT WPSRSEM-NP, REVISION 3 (TAC NO MB0306) dated September 10 2001.
Shutdown Margin	SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION ON "QUALIFICATIONS OF REACTOR PHYSICS METHODS FOR APPLICATION TO KEWAUNEE" REPORT, dated August 21, 1979, report date September 29, 1978
Moderator Temperature Coefficient	SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION ON "QUALIFICATIONS OF REACTOR PHYSICS METHODS FOR APPLICATION TO KEWAUNEE" REPORT, dated August 21, 1979, report date September 29, 1978
Shutdown Bank Insertion Limits	SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION ON "QUALIFICATIONS OF REACTOR PHYSICS METHODS FOR APPLICATION TO KEWAUNEE" REPORT, dated August 21, 1979, report date September 29, 1978 KEWAUNEE NUCLEAR POWER PLANT – REVIEW FOR KEWAUNEE RELOAD SAFETY EVALUATION METHODS TOPICAL REPORT WPSRSEM-NP, REVISION 3 (TAC NO MB0306) dated September 10 2001.

Relocated Parameter and Approved Methods

Parameter	NRC Approved Methodology
Control Bank Insertion Limits	SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION ON "QUALIFICATIONS OF REACTOR PHYSICS METHODS FOR APPLICATION TO KEWAUNEE" REPORT, dated August 21, 1979, report date September 29, 1978 KEWAUNEE NUCLEAR POWER PLANT – REVIEW FOR KEWAUNEE RELOAD SAFETY EVALUATION METHODS TOPICAL REPORT WPSRSEM-NP, REVISION 3 (TAC NO MB0306) dated September 10 2001.
Height Dependent Heat Flux Hot Channel Factor (F_Q)	SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION ON "QUALIFICATIONS OF REACTOR PHYSICS METHODS FOR APPLICATION TO KEWAUNEE" REPORT, dated August 21, 1979, report date September 29, 1978 Nissley, M.E. et, al., "Westinghouse Large-Break LOCA Best-Estimate Methodology," WCAP-10924-P-A, Volume 1, Revision 1, Addendum 4, March 1991, Volume 1: Model Description and Validation; Addendum 4: Model Revisions. N. Lee et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081-NP-A (Non-Proprietary), dated August 1985. C.M. Thompson, et al., "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condesation Model," WCAP-10054-P-A, Addendum 2, Revision 1 (Proprietary) and WCAP-10081-NP (Non-Proprietary), dated July 1997. ANF-88-133 (P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Meghodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation, dated

Relocated Parameter and Approved Methods	
Parameter	NRC Approved Methodology
	December 1991 XN-NF-77-57, Exxon Nuclear Power Distribution Control for Pressurized Water Reactors, Phase II, dated January 1978, and Supplement 2, dated October 1981.
Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)	SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION ON "QUALIFICATIONS OF REACTOR PHYSICS METHODS FOR APPLICATION TO KEWAUNEE" REPORT, Dated August 21, 1979, Report Date September 29, 1978 KEWAUNEE NUCLEAR POWER PLANT – REVIEW FOR KEWAUNEE RELOAD SAFETY EVALUATION METHODS TOPICAL REPORT WPSRSEM-NP, REVISION 3 (TAC NO MB0306) Dated September 10 2001. 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, dated October 1986. ANF-88-133 (P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Meghodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation, dated December 1991. EMF-92-116 (P)(A) Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Designs," Siemens Power Corporation, dated February 1999

Relocated Parameter and Approved Methods

Parameter	NRC Approved Methodology
Axial Flux Difference	SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION ON "QUALIFICATIONS OF REACTOR PHYSICS METHODS FOR APPLICATION TO KEWAUNEE" REPORT, dated August 21, 1979, report date September 29, 1978 KEWAUNEE NUCLEAR POWER PLANT – REVIEW FOR KEWAUNEE RELOAD SAFETY EVALUATION METHODS TOPICAL REPORT WPSRSEM-NP, REVISION 3 (TAC NO MB0306) dated September 10 2001.
Overtemperature ΔT	KEWAUNEE NUCLEAR POWER PLANT – REVIEW FOR KEWAUNEE RELOAD SAFETY EVALUATION METHODS TOPICAL REPORT WPSRSEM-NP, REVISION 3 (TAC NO MB0306) dated September 10 2001. WCAP-8745-P-A, "Design Basis For The Thermal Overpower Delta-T And Thermal Overtemperature Delta-T Trip Functions, September 1986"
Overpower ΔT	KEWAUNEE NUCLEAR POWER PLANT – REVIEW FOR KEWAUNEE RELOAD SAFETY EVALUATION METHODS TOPICAL REPORT WPSRSEM-NP, REVISION 3 (TAC NO MB0306) dated September 10 2001. WCAP-8745-P-A, "Design Basis For The Thermal Overpower Delta-T And Thermal Overtemperature Delta-T Trip Functions, September 1986"
DNB Limits	KEWAUNEE NUCLEAR POWER PLANT – REVIEW FOR KEWAUNEE RELOAD SAFETY EVALUATION METHODS TOPICAL REPORT WPSRSEM-NP, REVISION 3 (TAC NO MB0306) dated September 10 2001.

Relocated Parameter and Approved Methods	
Parameter	NRC Approved Methodology
Refueling Boron Concentration	SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION ON "QUALIFICATIONS OF REACTOR PHYSICS METHODS FOR APPLICATION TO KEWAUNEE" REPORT, dated August 21, 1979, report date September 29, 1978