From:	Feintuch, Karl
Sent:	Monday, January 30, 2012 5:33 PM
To:	Craig D Sly
Cc:	Jack Gadzala; Sun, Summer; Blumberg, Mark
Subject:	ME7110 Kewaunee Amendment Request Re: Chi-over-Q - RSXB Request for Additional Information (RAI) <ae>
Attachments:	response 9-12-11.pdf; Ref 5 ML030210062 03172003.pdf; Ref 6 ML040430633 02272004.pdf; Ref 7 ML070430020 03082007.pdf; Ref 1 NRC-11-066 COLR Cyc 31 R2 11-501 8-31-11.pdf; Ref 2 ML072290373 08302007.pdf; ME7110 Chi-over-Q RAI status Tracking 2012-02-28.xlsx

(DRAFT) REQUEST FOR ADDITIONAL INFORMATION
KEWAUNEE POWER STATION
LICENSE AMENDMENT REQUEST (TAC No. ME7110):
MODIFYING THE TECHNICAL SPECIFICATIONS (TS) AND
THE CURRENT LICENSING BASIS (CLB)
TO INCORPORATE CHANGES TO
THE CURRENT RADIOLOGICAL ACCIDENT ANALYSIS (RAA) OF RECORD
(KNOWN AS CHI-OVER-Q)
DOCKET NO. 50-305

By letter dated August 30, 2011, Dominion Energy Kewaunee (DEK) submitted a license amendment request (LAR)-244 (ADAMS Accession No. ML11252A521) to revise the Kewaunee Power Station (KPS) Operating License by modifying the Technical Specifications (TS) and the current licensing basis (CLB) to incorporate changes to the current radiological accident analysis (RAA) of record. This proposed amendment would revise the current RAA for the design-basis accidents (DBAs) described in Chapter 14 of the KPS Updated Safety Analysis Report (USAR). This amendment would also fulfill a commitment made to the NRC in response to Generic Letter 2003-01, “Control Room Habitability.”

In the course of their technical review, the Reactor Systems Technical Branch (SRXB) has requested further information items to enable completion of its support efforts for other branches. These items are provided in draft form for you to review for clarification. We seek to confirm your understanding of the items and the determination of a firm date for response, typically within 30 days of the date of this Request for Additional Information (RAI). The items we seek are listed below.

Please contact me by 02/01/2012 to confirm: (1) that the items are clear to you and to the responsive DEK staff without further discussion or (2) that a clarifying conference call is needed. Upon determination that the RAI items are clear and confirmation of when responses to these items are due, these draft RAI items will be considered to be in final form.

ME7110 is a complex project and we (Craig Sly of DEK and myself) have discussed methods for (1) improved movement of RAI information, (2) improved responsiveness to NRC staff requests, and (3) more flexibility for DEK to schedule RAI response activity, over that associated with more rigidly defined RAI milestone events. This group of two SRXB RAIs will be managed by the attached spreadsheet. This and subsequent RAI traffic will be tracked by an individual identifier to provide the associated response by the individualized “request by” date.

Docketing of this information by submittal under oath or affirmation will be managed by a reference to the associated ADAMS Accession No. (ML#) on the spreadsheet. Docketing will take place on groups of RAI item responses based on close completion schedules rather than close issuance schedules, as is now customary. Thus, if DEK can respond with individual information in 5 days, the “request by” date will be shortened and will

be received sooner than the rest of the items, although its docketing event might coincide with the original set of items or with those RAI items originating from another Technical Branch.

We will periodically assess when this new process is of mutual benefit while conforming to the regulations for processing amendment requests and their associated RAIs.

The two SRXB RAI items in this request are assigned the following tracking numbers. The associated entries are defined in the "Legend" tab of the spreadsheet:

1. ME7110-RAII-SRXB-Sun-001-2012-02-28
2. ME7110-RAII-SRXB-Sun-002-2012-02-28

1 - ME7110-RAII-SRXB-Sun-001-2012-02-28

In an email message from Craig Sly (DEK) to Karl Feintuch (USNRC) dated September 12, 2011 12:44 PM, DEK provided some information pertaining to the percentage (%) of "Failed Fuel Following the Accident." SRXB seeks to apply information contained in the file *"response 9-12-11.pdf"* (one among six pdf attachments to the email of September 12, 2011 12:44, all of which are included with this message, for completeness). SRXB is providing assistance to another Technical Branch rather than using this information to prepare a safety evaluation of the requested licensing action. In your RAI Response to this item, please provide file *"response 9-12-11.pdf."*

====================================

2 - ME7110-RAII-SRXB-Sun-002-2012-02-28

Application Attachment 4, page 154 indicates that the actuation time of the safety injection (SI) signal (in seconds) is changed from 52.5 to 240 during rod ejection accident (REA). The reason for the change, as stated by the licensee, is that the delay of the SI signal is conservative. The Current License Basis (CLB) assumption is based on a 2-inch diameter break. The REA is specified to have a smaller 1.6 inch diameter break. The SI signal generated from a 1-inch diameter break is 240 seconds.

It is not clear why a longer delay time of the actuation of the SI signal is conservative for the REA dose analysis.

Please provide justification of the longer SI actuation delay time used in the REA dose analysis.

===================================== end RAI Items requested ====================

Please contact me by February 1, 2012 if you have any questions or need to schedule a clarification conference call.

Karl Feintuch
USNRC
301-415-3079

========= begin email message sent September 12, 2011 12:44 PM from Sly (DEK, the licensee) to Karl Feintuch (USNRC), which contained 6 documents of which "*response 9-12-11.pdf*" is a subject of an RAI request in this message ===============

From: Craig D Sly [mailto:craig.d.sly@dom.com]
Sent: Monday, September 12, 2011 12:44 PM
To: Feintuch, Karl; Jack Gadzala
Subject: RE: Kewanee - radiological accident analysis (RAA)

Karl,

Let me start broadly and work down to the more specific question.

The questions posed during our telephone discussion are related to values located in Table 2.0-1 of the radiological accident analysis (RAA). The intent of Table 2.0-1 is to provide a summary of the design and licensing basis changes contained in the RAA. We were hopeful that this table would help the NRC staff in their review and also assist in generating an SER. But this Table is only a high

level summary. A complete discussion of the basis or reason for each of the changes cited in Table 2.0-1 is provided in Section 3 of the RAA.

> So for example: Table 2.0-1 (page 23 of RAA) contains three changes for the Locked Rotor Event. One of these changes is a decrease from 50% to 25% in "Failed Fuel Following the Accident." This is a high level summary of the change. Note that just above the Table row containing the "Failed Fuel Following the Accident" change is a Row stating "Locked Rotor Accident (Section 3.6)." This refers the reader to section 3.6 of the RAA for details concerning the LRA radiological release analysis.
>
> Section 3.6 of the RAA starts on page 127. This section provides a scenario description, source term definition, etc. On page 131 a schematic of the rad release is provided. Starting on page 132, Table 3.6-1 provides Basic Data and Assumptions for the LRA. This Table has columns for Parameter, Current License Basis Values, Proposed new CLB Values, and Reason for Change. For those values that are changing, a short discussion is provided in the "Reason for Change" column. The reason cited for the 50% to 25% change above is as follows: "Rods-in-DNB analysis show approximately 7% rods-in DNB following a LRA for the current cycle. 25% is specified in the reload safety analysis checklist (RSAC)."

Concerning the specific question related to the change from 50% to 25% in "Failed Fuel Following the Accident." A response is provided in the attached file labeled "response 9-12-11".   I've also attached a copy of several of the references used in this response for convenience.

Let me know if you have any questions related to this.

Thanks,

Craig Sly
Dominion Resources Services, Inc.
Nuclear Licensing and Operations Support
W: 804-273-2784
C: 804-241-2473

Craig

From: Feintuch, Karl [mailto:Karl.Feintuch@nrc.gov]
Sent: Thursday, September 08, 2011 2:08 PM
To: Craig D Sly (Generation - 6); Jack Gadzala (Generation - 4)
Subject: FW: Kewanee - radiological accident analysis (RAA)

I need a conference call for this.

From: Sun, Summer
Sent: Thursday, September 08, 2011 11:05 AM
To: Feintuch, Karl
Cc: Ulses, Anthony
Subject: Kewanee - radiological accident analysis (RAA)

Karl, I have the following question to discuss with the licensee before I could decide the information provided in August letter is sufficient for us to continue the review. Please arrange a conference for me with the licensee. I'll be available this afternoon and next Monday. Thanks.

Issue to be clarified-

Page 86 of attachment 4 indicates that the radiological accident analysis (RAA) for the steam generator tube rupture (SGTR) is based on an analysis with break flow continuing for 55 minutes.

Please clarify if SGTR analysis for 55 minutes is the current design basis or not. If it is, please refer specific page of the NRC SE approving the analysis. If not, address acceptability of the analysis.

Also, confirm that the thermal hydraulic and system response analyses used to support the RAA are the same analyses documented in the USAR for the LOCA, Locked Rotor, Rod Ejection, and Steam Line Break events.

========== end email message sent September 12, 2011 12:44 PM from Sly (DEK, the licensee) to Karl Feintuch (USNRC), which contained 6 documents of which "*response 9-12-11.pdf*" is a subject of an RAI request in this message ===============

References:

1. Letter from M. J. Wilson (DEK) to Document Control Desk, “Core Operating Limits Report Cycle 31 Revision 2,” dated August 31, 2011.

2. Letter from P. D. Milano (NRC) to David A. Christian (DEK), “Kewaunee Power Station – Safety Evaluation for Topical Report DOM-NAF-5 (TAC NO. MD2829),” dated Aug. 30, 2007.

3. KPS USAR Revision 22.06, updated 6/30/2011

4. Letter from J. A. Price (DEK) to Document Control Desk, “License Amendment Request 244: Proposed Revision to Radiological Accident Analysis and Control; Room Envelope Habitability Technical Specifications,” dated August 30, 2011 [ML 112520670].

5. Letter from John Lamb (NRC) to Tom Coutu (NMC), “Kewaunee Nuclear Power Plant – Issuance of Amendment Regarding Implementation of Alternate Source Term (TAC No. MB4596),” dated March 17, 2003. [ML030210062]

6. Letter from John Lamb (NRC) to Tom Coutu (NMC), Kewaunee Nuclear Power Plant – Issuance of Amendment Regarding Stretch Power Uprate (TAC No. MB9031),” dated February 27, 2004. [ML040430633]

7. Letter from R. F. Kuntz (NRC) to D. A. Christian (DEK), “Kewaunee Power Station – Issuance of Amendment RE: Radiological Accident Analysis and Associated Technical Specification Change (TAC No. MC9715),” dated March 8, 2007. [ML070430020]

<u>Approved Safety Analysis and Reload Safety Evaluation Methods</u>

Dominion Topical Report DOM-NAF-5-A “Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS)” (approved in Reference 2) is included in the list of approved methodologies in the KPS COLR (Reference 1) (see page 16 of 17, item 1) and also in the KPS TS 5.6.3 (see TS page 5.6-3, item 15). Per NRC approval of DOM-NAF-5-A, Dominion is allowed to use its reload topical (VEP-FRD-42-A), core design topical (DOM-NAF-1-A), statistical DNB topical (VEP-NE-2-A), RPDC topical (VEP-NE-1-A), core thermal-hydraulics topical (DOM-NAF-2-A) and RETRAN topical (VEP-FRD-41-A) for evaluation of reload cores at KPS (Note these Dominion topical reports are referenced within the Dominion topical report DOM-NAF-5-A).

Dominion Topical Report VEP-FRD-42-A describes the application of a bounding parameters approach to evaluating Dominion reload cores. Under this topical, a reload core is evaluated against a bounding analysis by a process of parameter comparison. For a proposed core design, if all key analysis parameters are conservatively bounded, then the reference safety analysis is assumed to apply and no further analysis is necessary. Transient analyses are performed using RETRAN models developed under the method approved in Dominion Topical Report VEP-FRD-41-A and core thermal-hydraulics are evaluated under Dominion Topical Report DOM-NAF-2-A. Dominion Topical Report DOM-NAF-5-A, Attachment B Section 4.2, discusses the Locked Rotor event in depth and includes a benchmark of the Locked Rotor analysis using Dominion methods to the previously existing USAR analysis.

Approved Radiological Accident Analysis Methods

Reference 4 (see page 7/191 of attachment 4) describes the current KPS analyses of record for radiological events (including the locked rotor event) that were previously approved in KPS Amendment No. 166 (Reference 5), which implemented the Alternate Source Term (AST); and Amendment No. 172 (Reference 6), which implemented a stretch power uprate to 1772 megawatts thermal (MWt); and Amendment No. 190 (Reference 7), which implemented the radiological analyses with higher control room emergency zone (CREZ) unfiltered in-leakage. The current locked rotor radiological analysis methods were approved as part of References 5, 6, and 7. The approved radiological analyses support the locked rotor failed fuel limit of 50% described in Reference 3, Section 14.1.8.

Dominion radiological accident analyses in Reference 4 support a new locked rotor failed fuel limit of 25%. As described in Reference 4 (see Table 3.6-1 on page 132 of 191) the reload safety analysis checklist (RSAC) will incorporate the failed fuel limit from the Dominion radiological accident analysis as a design criterion for the reload core design. The calculated percent (%) rods- in- DNB (fuel in DNB is assumed to fail for the purposes of calculating a radiological release) must be below the dose analysis % failed fuel limit on a reload basis. The 25% failed fuel limit is supported with information from reload calculations that show rods-in-DNB following a LRA for the current fuel cycle is well below the 25% limit (see Reference 4, Table 3.6-1, page 132 of 191).

Conclusion:

LAR 244 requests NRC review and approval of the locked rotor dose consequences analysis, which includes a revised assumption for failed fuel rod limit of 25% (decreased from the current analysis assumption of 50%). The thermal-hydraulic analysis for the locked rotor event, which was performed with the NRC-approved methods of Reference 2, has not been revised for LAR 244. Future reload analysis of the locked rotor event will continue to be

performed in accordance with the NRC-approved methodology in DOM-NAF-5-A (Reference 2).

Dominion Energy Kewaunee, Inc.
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Dominion®

AUG 31 2011

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Serial No. 11-501
LIC/NW/R0
Docket No.: 50-305
License No.: DPR-43

DOMINION ENERGY KEWAUNEE, INC.
KEWAUNEE POWER STATION
CORE OPERATING LIMITS REPORT CYCLE 31 REVISION 2

Pursuant to Kewaunee Power Station (KPS) Technical Specification 5.6.3.d, enclosed is a copy of the Kewaunee Power Station Core Operating Limits Report Cycle 31, Revision 2.

If you have questions or require additional information, please feel free to contact Mr. Jack Gadzala at 920-388-8604.

Very truly yours,

for

Michael J. Wilson
Director Safety and Licensing
Kewaunee Power Station

Commitments made by this letter: NONE

Enclosure
1. Kewaunee Power Station Core Operating Limits Report Cycle 31, Revision 2.

cc: Regional Administrator, Region III
U. S. Nuclear Regulatory Commission
2443 Warrenville Road
Suite 210
Lisle, IL 60532-4352

Mr. K. D. Feintuch
Project Manager
U.S. Nuclear Regulatory Commission
One White Flint North, Mail Stop O8-H4A
11555 Rockville Pike
Rockville, MD 20852-2738

NRC Senior Resident Inspector
Kewaunee Power Station

CORE OPERATING LIMITS REPORT
Kewaunee Unit 1 Cycle 31
Revision 2

August 2011

1.0 INTRODUCTION

This Core Operating Limits Report (COLR) for Kewaunee Unit 1 Cycle 31 has been prepared in accordance with the requirements of Kewaunee Technical Specification 5.6.3.

A cross reference between the COLR section and the KPS Technical Specifications affected by this report is given below:

COLR Section	KPS Technical Specification	Description
2.1	2.1.1	Reactor Core Safety Limit
2.2	3.1.1	Shutdown Margin
2.3	3.1.3	Moderator Temperature Coefficient (MTC)
2.4	3.1.5	Shutdown Bank Insertion Limits
2.5	3.1.6	Control Bank Insertion Limits
2.6	3.2.1	Heat Flux Hot Channel Factor ($F_Q(Z)$)
2.7	3.2.2	Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$)
2.8	3.2.3	AXIAL FLUX DIFFERENCE (AFD)
2.9	3.3.1 Function 6	Reactor Protection System Instrumentation: Overtemperature ΔT
2.10	3.3.1 Function 7	Reactor Protection System Instrumentation: Overpower ΔT
2.11	3.4.1	RCS Pressure, Temperature and Flow Departure from Nucleate Boiling (DNB) Limits
2.12	3.9.1	Boron Concentration (Refueling Operations)
Figure 1	2.1	Reactor Core Safety Limits Curve (1772 MWt)
Figure 2		DELETED (Required Shutdown Margin)
Figure 3		DELETED (Hot Channel Factor Normalized Operating Envelope (K(Z)))
Figure 4	3.1.6	Control Bank Insertion Limits
Figure 5		N(Z) Values (Top and Bottom 9% excluded)
Figure 6		DELETED (Penalty Factor, Fp, for $F_Q(Z)$)
Figure 7	3.2.3	AXIAL FLUX DIFFERENCE Envelope

2.0 <u>OPERATING LIMITS</u>

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. These limits have been developed using the NRC-approved methodologies specified in Technical Specification 5.6.3.

2.1 <u>Reactor Core Safety Limits</u> (TS 2.1.1)

The combination of rated power level, coolant pressure, and coolant temperature shall not exceed the limits shown in COLR Figure 1 (1772 MWt). The safety limit is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line.

2.2 <u>Shutdown Margin</u> (TS 3.1.1)

Shutdown Margin shall be ≥ **1554 pcm**.

2.3 <u>Moderator Temperature Coefficient (MTC)</u> (TS 3.1.3)

2.3.1 When the reactor is critical and ≤ 60% of RATED THERMAL POWER, the moderator temperature coefficient shall be ≤ 5.0 pcm/°F. When the reactor is > 60% RATED THERMAL POWER, the moderator temperature coefficient shall be zero or negative.

a. The BOC/ARO-MTC shall be ≤ 5.0 pcm/°F (upper limit), when ≤ 60% RTP, and ≤ 0.0 pcm/°F when > 60% RTP.

b. The EOC/ARO/RTP-MTC shall be less negative than or equal to **-44.7** pcm/°F (lower limit).

MTC surveillance limits are:

i) The 300 ppm/ARO/RTP-MTC should be less negative than or equal to **-39.1 pcm/°F**. If MTC is more negative, then repeat measurement once per 14 EFPD during the remainder of the fuel cycle. Note this surveillance does not need to be repeated if criterion ii, listed below, is satisfied.

ii) The 60 ppm/ARO/RTP-MTC should be less negative than or equal to **-43.6 pcm/°F**.

2.4 <u>Shutdown Bank Insertion Limits</u> (TS 3.1.5)

The shutdown rods shall be fully withdrawn (224 steps) when the reactor is critical or approaching criticality.

2.5 <u>Control Bank Insertion Limits</u> (TS 3.1.6)

The control rod banks shall be limited in physical insertion as shown in **COLR Figure 4**.

2.6 <u>Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$)</u> (TS 3.2.1)

2.6.1 $F_Q^C(Z)$ Limits for Fuel

$$F_Q^N(Z) * 1.03 * 1.05 \leq \frac{CFQ}{P} * K(Z) \qquad \textit{for}\ P > 0.5 \qquad [422V+]$$

$$F_Q^N(Z) * 1.03 * 1.05 \leq \frac{CFQ}{0.5} * K(Z) \qquad \textit{for}\ P \leq 0.5 \qquad [422V+]$$

Where:

P	is the fraction of full power at which the core is operating
K(Z)	is 1.0 for all core heights
Z	is the core height location for the FQ of interest
CFQ	**equals 2.50**
$F_Q^N(Z)$	is a measured FQ distribution obtained during the target flux determination

2.6.2 $F_Q^T(Z)$ Limits for Fuel

$$F_Q^N(Z) * 1.03 * 1.05 * N(Z) * F_P \leq \frac{CFQ}{P} * K(Z) \qquad [422V+]$$

Where:

P	is the fraction of full power at which the core is operating
K(Z)	is 1.0 for all core heights
Z	is the core height location for the FQ of interest
CFQ	**equals 2.50**
F_p	is the penalty factor described in 2.6.3
N(Z)	is a cycle-specific non-equilibrium multiplier on $F_Q^N(Z)$ to account for power distribution transients during normal operation, provided in Figure 5.
$F_Q^N(Z)$	is a measured FQ distribution obtained during the target flux determination

The N(z) decks are generated for normal operation flux maps that are typically taken at full power, ARO. Additional N(z) decks may be generated, if necessary, consistent with the methodology described in Reference 1.

2.6.3 A penalty factor of 1.00 shall be used unless the Note criteria of TS SR 3.2.1.2 is met, at which time a penalty of 1.02 shall be used.

2.7 <u>Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)</u> (TS 3.2.2)

$$F^N_{\Delta H} * 1.04 \leq CFDH * [1 + PFDH * (1 - P)] \qquad [422\ V+]$$

Where:

P is the fraction of full power at which the core is operating
CFDH equals 1.70
PFDH equals 0.3

2.8 <u>AXIAL FLUX DIFFERENCE (AFD)</u> (TS 3.2.3)

The AFD acceptable operation limits are provided in **COLR Figure 7**.

2.9 <u>Overtemperature ΔT Setpoint</u> (TS 3.3.1 Function 6)

$$\Delta T \leq \Delta T_0 * \left[K_1 - K_2 * (T - T') * \frac{1 + \tau_1 s}{1 + \tau_2 s} + K_3 * (P - P') - f_1(\Delta I) \right]$$

ΔT_0 = Indicated ΔT at RATED THERMAL POWER, %
s = Laplace transform operator, sec^{-1}.
T = Average temperature, °F
T' ≤ **573.0 °F**
P = Pressurizer Pressure, psig
P' ≥ **2235 psig**
K_1 ≤ **1.195**
K_2 ≥ **0.015/°F**
K_3 ≥ **0.00072/psig**
τ_1 ≥ **30 seconds**
τ_2 ≤ **4 seconds**
$f_1(\Delta I)$ = An even function of the indicated difference between top and bottom detectors of the power range nuclear ion chambers. Selected gains are based on measured instrument response during plant startup tests, where qt and qb are the percent power in the top and bottom halves of the core respectively and qt + qb is total core power in percent RATED THERMAL POWER, such that

(a) For qt - qb within **-13.5, +4.5** %, f(ΔI) = 0
(b) For each percent that the magnitude of qt - qb exceeds **+4.5%**, the ΔT trip setpoint shall be automatically reduced by an equivalent of **1.51%** of RATED THERMAL POWER.
(c) For each percent that the magnitude of qt - qb exceeds **-13.5%**, the ΔT trip setpoint shall be automatically reduced by an equivalent of **3.78%** of RATED THERMAL POWER.

2.10 Overpower ΔT Setpoint (TS 3.3.1 Function 7)

$$\Delta T \leq \Delta T_0 * \left[K_4 - K_5 * \frac{\tau_3 s}{\tau_3 s + 1} * T - K_6 * (T - T') - f_2(\Delta I) \right]$$

ΔT_0	=	Indicated ΔT at RATED THERMAL POWER, %
s	=	Laplace transform operator, sec^{-1}.
T	=	Average temperature, °F
T'	≤	**573.0 °F**
K_4	≤	**1.095**
K_5	≥	**0.0275/°F** for increasing T
	≥	**0** for decreasing T
K_6	≥	**0.00103/°F** for T > T'
	≥	**0** for T < T'
τ_3	≥	**10 seconds**
$f_2(\Delta I)$	=	0 for all ΔI

2.11 RCS Pressure, Temperature and Flow Departure from Nucleate Boiling (DNB) Limits (TS 3.4.1)

2.11.1 During steady state power operation, Tavg shall be ≤ **576.7 °F** for control board indication or ≤ **576.5 °F** for computer indication.

2.11.2 During steady state power operation, pressurizer pressure shall be ≥ **2217 psig** for control board indication or ≥ **2219 psig** for computer indication.

2.11.3 During steady state power operation, reactor coolant total flow rate shall be ≥ **186,000 gpm.**

2.12 Boron Concentration (Refueling Operations) (TS 3.9.1)

When there is fuel in the reactor, a minimum boron concentration of **2500** 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.

Figure 1
(TS 2.1.1)
Reactor Core Safety Limits Curve (1772 Mwt)
(Cores Containing 422V+ fuel)

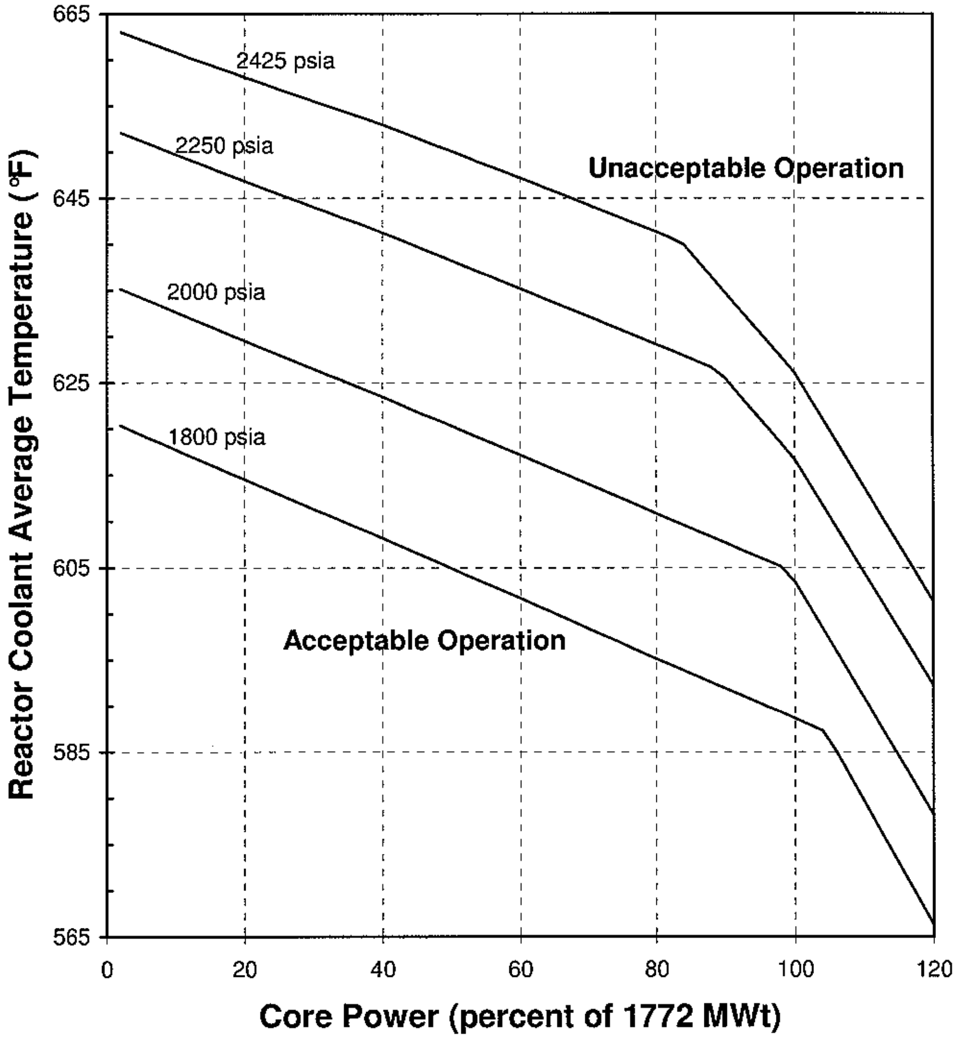


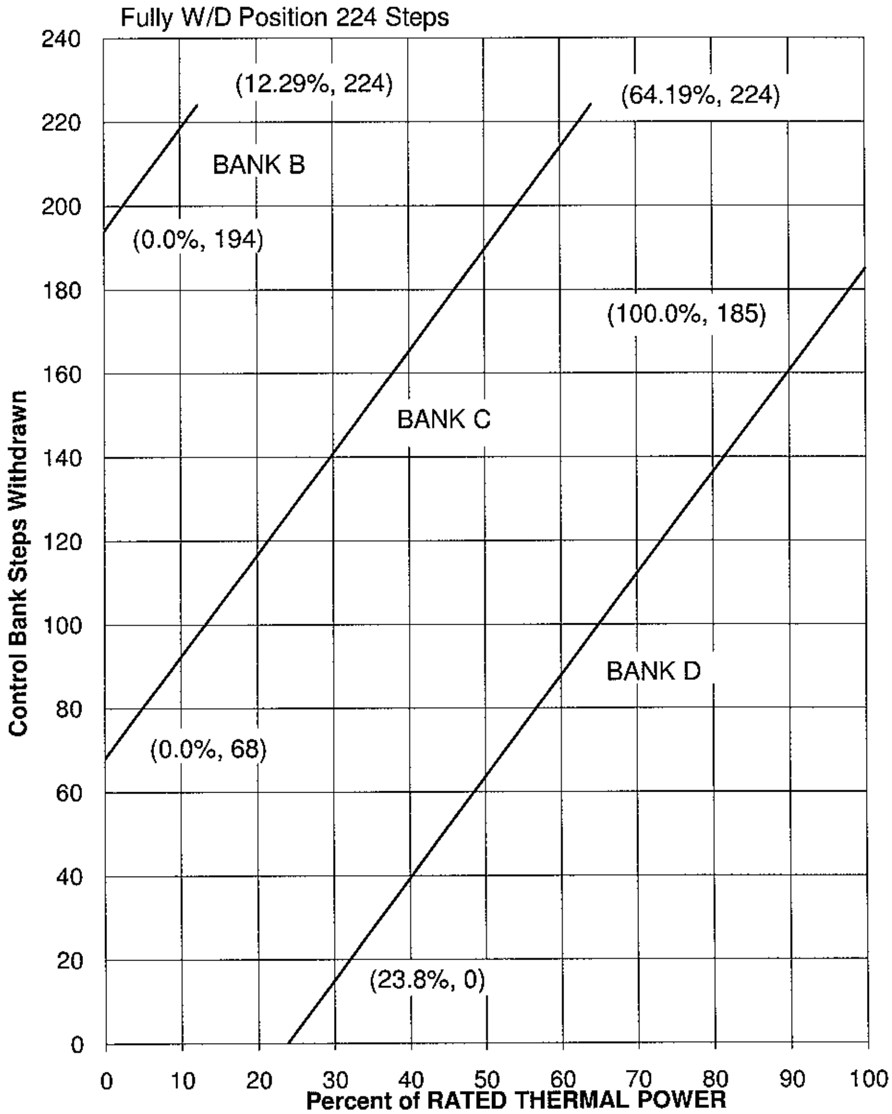
Figure 2
Required Shutdown Margin vs. Boron Concentration

DELETED

Figure 3
Hot Channel Factor Normalized Operating Envelope (K(Z))

DELETED

Figure 4
Control Bank Insertion Limits
(TS 3.1.6)



Note: The Rod Bank Insertion Limits are based on a control bank tip-to-tip distance of 126 steps.

Figure 5
N(Z) Values [1]

NODE	HEIGHT (FEET)	0 to 1000 MWD/MTU	1000 to 3000 MWD/MTU	3000 to 5000 MWD/MTU	5000 to 7000 MWD/MTU	7000 to 9000 MWD/MTU	9000 to 11000 MWD/MTU
		AO = 2.3%	AO = 0.8%	AO = -1.0%	AO = -2.0%	AO = -2.4%	AO = -2.6%
Top							
7	10.8	1.100	1.110	1.118	1.146	1.148	1.157
8	10.6	1.099	1.111	1.119	1.144	1.145	1.154
9	10.4	1.100	1.108	1.116	1.141	1.142	1.151
10	10.2	1.099	1.105	1.116	1.138	1.139	1.148
11	10.0	1.096	1.102	1.120	1.135	1.136	1.145
12	9.8	1.092	1.097	1.123	1.130	1.132	1.143
13	9.6	1.089	1.096	1.126	1.127	1.129	1.142
14	9.4	1.084	1.099	1.126	1.126	1.127	1.138
15	9.2	1.083	1.106	1.128	1.132	1.132	1.137
16	9.0	1.084	1.113	1.136	1.144	1.143	1.140
17	8.8	1.089	1.119	1.147	1.157	1.157	1.148
18	8.6	1.095	1.125	1.153	1.166	1.165	1.159
19	8.4	1.102	1.128	1.157	1.171	1.170	1.170
20	8.2	1.108	1.131	1.160	1.175	1.175	1.175
21	8.0	1.112	1.132	1.160	1.177	1.179	1.179
22	7.8	1.117	1.131	1.160	1.178	1.184	1.184
23	7.6	1.120	1.129	1.158	1.178	1.188	1.188
24	7.4	1.122	1.127	1.157	1.178	1.190	1.190
25	7.2	1.124	1.125	1.156	1.176	1.192	1.192
26	7.0	1.124	1.124	1.152	1.173	1.192	1.192
27	6.8	1.123	1.122	1.139	1.168	1.188	1.188
28	6.6	1.120	1.120	1.123	1.161	1.181	1.181
29	6.4	1.118	1.118	1.114	1.157	1.178	1.178
30	6.2	1.115	1.115	1.112	1.154	1.174	1.174
31	6.0	1.113	1.112	1.108	1.149	1.168	1.168
32	5.8	1.109	1.109	1.106	1.141	1.159	1.159
33	5.6	1.102	1.102	1.102	1.130	1.148	1.147
34	5.4	1.099	1.099	1.099	1.121	1.134	1.134
35	5.2	1.102	1.102	1.095	1.114	1.119	1.121
36	5.0	1.107	1.107	1.092	1.111	1.110	1.111
37	4.8	1.111	1.111	1.088	1.110	1.109	1.105
38	4.6	1.120	1.119	1.091	1.107	1.107	1.106
39	4.4	1.131	1.129	1.100	1.101	1.101	1.111
40	4.2	1.136	1.134	1.107	1.100	1.094	1.114
41	4.0	1.136	1.134	1.110	1.105	1.091	1.114
42	3.8	1.137	1.135	1.114	1.108	1.093	1.112
43	3.6	1.143	1.141	1.119	1.110	1.101	1.110
44	3.4	1.153	1.150	1.123	1.111	1.107	1.109
45	3.2	1.167	1.163	1.129	1.113	1.113	1.110
46	3.0	1.181	1.176	1.135	1.117	1.118	1.115
47	2.8	1.195	1.189	1.144	1.122	1.122	1.120
48	2.6	1.208	1.201	1.153	1.123	1.123	1.121

Figure 5 (continued)
N(Z) Values [1]

NODE	HEIGHT (FEET)	0 to 1000 MWD/MTU	1000 to 3000 MWD/MTU	3000 to 5000 MWD/MTU	5000 to 7000 MWD/MTU	7000 to 9000 MWD/MTU	9000 to 11000 MWD/MTU
49	2.4	1.222	1.214	1.164	1.126	1.126	1.124
50	2.2	1.240	1.230	1.179	1.135	1.135	1.134
51	2.0	1.251	1.241	1.188	1.142	1.142	1.141
52	1.8	1.256	1.245	1.191	1.143	1.143	1.143
53	1.6	1.260	1.249	1.195	1.145	1.145	1.145
54	1.4	1.268	1.257	1.202	1.150	1.150	1.150
55	1.2	1.275	1.263	1.207	1.154	1.154	1.155
Bottom							

1) Excludes top and bottom 9%

These decks were generated for normal operation flux maps that are typically taken at full power ARO. Additional N(z) decks may be generated, if necessary, consistent with the methodology described in Reference 1.

Figure 5 (continued)
N(Z) Values [1]

NODE	HEIGHT (FEET)	11000 to 13000 MWD/MTU	13000 MWD/MTU to EOR
		AO = -2.7%	AO = -2.7%
Top			
7	10.8	1.157	1.158
8	10.6	1.154	1.155
9	10.4	1.151	1.153
10	10.2	1.148	1.151
11	10.0	1.145	1.150
12	9.8	1.143	1.148
13	9.6	1.142	1.146
14	9.4	1.137	1.140
15	9.2	1.137	1.139
16	9.0	1.144	1.145
17	8.8	1.158	1.158
18	8.6	1.171	1.171
19	8.4	1.181	1.181
20	8.2	1.187	1.187
21	8.0	1.190	1.190
22	7.8	1.193	1.193
23	7.6	1.196	1.195
24	7.4	1.197	1.196
25	7.2	1.198	1.198
26	7.0	1.195	1.197
27	6.8	1.186	1.192
28	6.6	1.173	1.184
29	6.4	1.168	1.181
30	6.2	1.164	1.177
31	6.0	1.157	1.172
32	5.8	1.148	1.162
33	5.6	1.140	1.151
34	5.4	1.131	1.137
35	5.2	1.120	1.121
36	5.0	1.111	1.108
37	4.8	1.106	1.102
38	4.6	1.106	1.097
39	4.4	1.111	1.090
40	4.2	1.114	1.085
41	4.0	1.114	1.084
42	3.8	1.112	1.084
43	3.6	1.110	1.087
44	3.4	1.109	1.091
45	3.2	1.110	1.099
46	3.0	1.114	1.108
47	2.8	1.120	1.117
48	2.6	1.120	1.120

Figure 5 (continued)
N(Z) Values [1]

NODE	HEIGHT (FEET)	11000 to 13000 MWD/MTU	13000 MWD/MTU to EOR
49	2.4	1.125	1.125
50	2.2	1.138	1.138
51	2.0	1.149	1.149
52	1.8	1.152	1.152
53	1.6	1.155	1.155
54	1.4	1.159	1.159
55	1.2	1.163	1.163
Bottom			

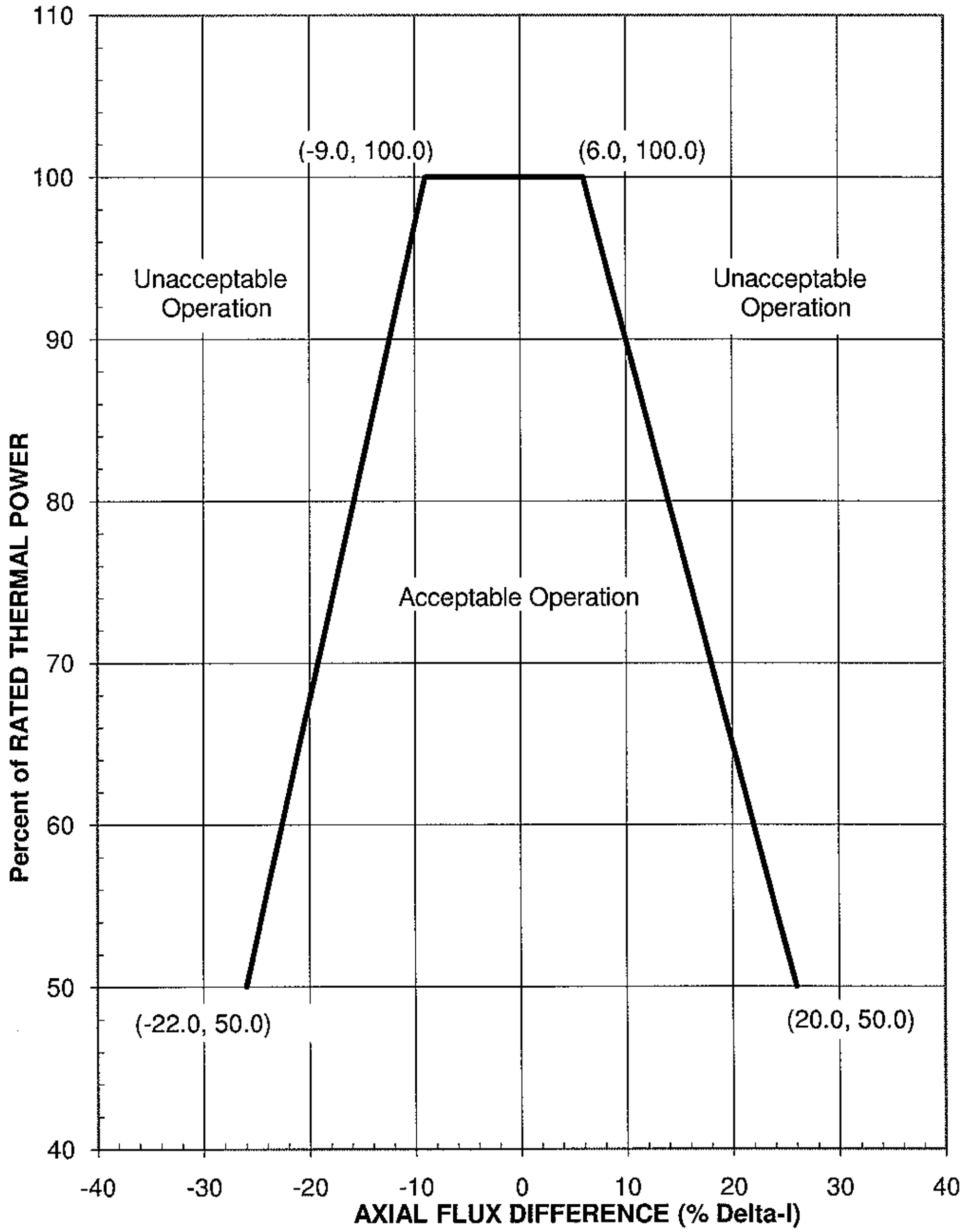
1) Excludes top and bottom 9%

These decks were generated for normal operation flux maps that are typically taken at full power ARO. Additional N(z) decks may be generated, if necessary, consistent with the methodology described in Reference 1.

Figure 6
Penalty Factor, Fp, for F_Q (Z)

DELETED

Figure 7
AXIAL FLUX DIFFERENCE Target Band
(TS 3.2.3)



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1. Topical Report DOM-NAF-5-A, Revision 0.2-A, "Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS)," January 2011.

   Methodology for:
   TS 2.1.1 – Reactor Core Safety Limit;
   TS 3.1.1 – Shutdown Margin;
   TS 3.1.3 – Moderator Temperature Coefficient;
   TS 3.1.5 – Shutdown Bank Insertion Limits;
   TS 3.1.6 – Control Bank Insertion Limits;
   TS 3.2.1 – Heat Flux Hot Channel Factor ($F_Q(Z)$);
   TS 3.2.2 – Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^{N}$);
   TS 3.2.3 – AXIAL FLUX DIFFERENCE (AFD);
   TS 3.4.1 – RCS Pressure, Temperature and Flow Departure from Nucleate Boiling (DNB) Limits;
   TS 3.9.1 –Boron Concentration (Refueling Operations)

2. Topical Report WPSRSEM-NP, Revision 3, "Kewaunee Nuclear Power Plant – Review for Kewaunee Reload Safety Evaluation Methods," September 10, 2001.

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   TS 3.1.1 – Shutdown Margin

3. WCAP-12945-P-A (Proprietary), "Westinghouse Code Qualification Document for Best-Estimate Loss-of-Coolant Accident Analysis," Volume I, Revision 2, and Volume II-V, Revision 1, March 1998.

   Methodology for:
   TS 3.2.1 – Heat Flux Hot Channel Factor ($F_Q(Z)$)
   TS 3.2.2 – Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^{N}$);

4. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.

   Methodology for:
   TS 3.2.1 – Heat Flux Hot Channel Factor ($F_Q(Z)$)

5. WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and the COSI Condensation Model," July 1997.

   Methodology for:
   TS 3.2.1 – Heat Flux Hot Channel Factor ($F_Q(Z)$)

6. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

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   TS 2.1.1 – Reactor Core Safety Limit;
   TS 3.1.3 – Moderator Temperature Coefficient;

7. WCAP-8745-P-A, "Design Bases for the Thermal Overtemperature ΔT and Thermal Overpower ΔT trip functions," September 1986.

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   TS 3.3.1 Function 6 – Overtemperature ΔT Setpoint;
   TS 3.3.1 Function 7 – Overpower ΔT Setpoint

8. WCAP-14449-P-A, Revision 1, "Application of Best Estimate Large-Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection," October 1999.

   Methodology for:
   TS 3.2.1 – Heat Flux Hot Channel Factor ($F_Q(Z)$)
   TS 3.2.2 – Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$);

9. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995.

   Methodology for:
   TS 3.2.1 – Heat Flux Hot Channel Factor ($F_Q(Z)$)

10. CENP-397-P-A, Revision 1, "Improved Flow Measurement Accuracy Using Cross Flow Ultrasonic Flow Measurement Technology," May 2000.

    Methodology for:
    TS 3.3.1 Function 6 – Overtemperature ΔT Setpoint;
    TS 3.3.1 Function 7 – Overpower ΔT Setpoint

August 30, 2007

Mr. David A. Christian
President and Chief Nuclear Officer
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: KEWAUNEE POWER STATION - SAFETY EVALUATION FOR TOPICAL REPORT DOM-NAF-5 (TAC NO. MD2829)

Dear Mr. Christian:

On August 16, 2006, as supplemented on December 6, 2006, April 16, May 4, and June 12, 2007, Dominion Energy Kewaunee submitted Topical Report (TR) DOM-NAF-5, "Application of Dominion Nuclear Core Design and Safety Analysis methods to Kewaunee Power Station (KPS)."

The Nuclear Regulatory Commission (NRC) staff has found that DOM-NAF-5 is acceptable for referencing in licensing applications for KPS to the extent specified and under the limitations delineated in the TR and in the enclosed safety evaluation (SE). The SE defines the basis for acceptance of the TR.

The NRC staff's acceptance applies only to material provided in the subject TR. The staff does not intend to repeat its review of the acceptable material described in the TR. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards. If future changes to the NRC's regulatory requirements affect the acceptability of this TR, Dominion Energy Kewauee will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

/RA/

Patrick D. Milano, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-305

Enclosure:
Safety Evaluation

cc w/encls: See next page

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SUBJECT: KEWAUNEE POWER STATION - SAFETY EVALUATION FOR TOPICAL REPORT DOM-NAF-5 (TAC NO. MD2829)

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Sincerely,

/RA/

Patrick D. Milano, Senior Project Manager
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Division of Operating Reactor Licensing
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Docket No. 50-305
Enclosure:
Safety Evaluation
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO FACILITY OPERATING LICENSE NO. DPR-43

DOMINION ENERGY KEWAUNEE, INC.

KEWAUNEE POWER STATION

DOCKET NO. 50-305

1.0 INTRODUCTION

By application dated August 16, 2006 (Reference 1), as supplemented on December 6, 2006, April 16, May 4, and June 12, 2007 (Agencywide Documents Access Management System (ADAMS) Accession Nos. ML070120088, ML063410177, ML071060392, ML071270780, and ML071630521, respectively), Dominion Energy Kewaunee, Inc. (the licensee) requested approval of Topical Report DOM-NAF-5, "Application Of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS)." The report describes the various in-scope design and analysis methodologies and documents the assessments of the applicability of these methodologies to KPS. The approval by the Nuclear Regulatory Commission (NRC) would permit the licensee to subsequently request an amendment to the Technical Specifications (TSs) to apply the Dominion Energy (Dominion) nuclear core design and safety analysis methods to the KPS design and licensing analyses.

The nuclear core design methods addressed by the report include the Reload Nuclear Design Methodology, Relaxed Power Distribution Control (RPDC) Methodology, and the Studsvik Core Management System (CMS) Reactor Physics Methods. The safety analysis methods covered by the report include the Vepco Reactor System Transient Analyses using the RETRAN Computer Code, Statistical Departure from Nucleate Boiling ration (DNBR) Evaluation Methodology, and the Reactor Core Thermal-Hydraulics using the VIPRE-D Computer Code. Attachments A (Reference 2) and B (Reference 3) to DOM-NAF-5 provide supplemental material documenting the applicability of Studsvik CMS Reactor Physics methods and Dominion's RETRAN methods to KPS.

2.0 REGULATORY EVALUATION

The NRC staff used the following requirements and guidance documents in evaluating the licensee's amendment request:

Section 50.34, "Contents of Applications; Technical Information," of Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR) requires that Safety Analysis Reports analyze the design and performance of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents.

As part of the core reload process, licensees perform reload safety evaluations to ensure that their safety analyses remain bounding for the design cycle. To confirm that the analyses remain bounding, they confirm that the inputs to the safety analyses are conservative with respect to the current design cycle. These inputs are checked using analytical models, and if key safety analysis parameters are not bounded, further analysis of the affected transients or accidents is performed to ensure that the applicable acceptance criteria are satisfied.

3.0 TECHNICAL EVALUATION OF ANALYTICAL METHODS AND APPLICABILITY

The core reload design and safety analysis process is currently performed by the KPS fuel supplier (Westinghouse), whereas other Dominion nuclear plants rely on the Dominion nuclear core design and safety analysis methods. In TR DOM-NAF-5, the licensee has proposed to apply the Dominion nuclear core design and safety analysis methods to KPS, although the KPS fuel supplier will continue to license the fuel design, perform fuel rod design analysis for reload fuel performance assessment and perform certain specific safety analyses, such as small-break and large-break loss-of-coolant accident (LOCA) analyses. The Dominion methods detailed herein are to be applied to KPS in a manner consistent with the conditions and limitations of this safety evaluation report (SER), other relevant NRC SERs and the relevant Dominion TRs.

3.1 Reload Nuclear Design Methodology

The reload nuclear design methodology in Dominion TR VEP-FRD-42, Revision 2.0-A, “Reload Nuclear Design Methodology” (Reference 6), consists of the analytical models, methods, reload design and reload safety analysis, and an overview of analyzed accidents. It is an iterative process that involves the determination of a core loading pattern that fulfills cycle energy requirements and the demonstration that the plant with the reload core satisfies the constraints of the plant design basis and safety analysis limits.

The reload safety evaluation uses a bounding analysis concept in which key analysis parameters with limiting directions are identified such that, if all key analysis parameters are conservatively bounded, a reference safety analysis is applicable and no further analysis is necessary. If any values are not bounded, further analysis of the transient or accident in question is performed, the applicable safety analyses are revised, or changes are made in the operating requirements to satisfy applicable safety analysis criteria. The safety analysis process typically consists of steady state nuclear calculations used to derive the core physics related key analysis parameters as well as a dynamic accident analysis that utilizes these parameters to determine the accident result.

The Dominion nuclear design methodology and the current KPS reload design methodology are similar and share a common basis in Westinghouse TR WCAP-9272, “Westinghouse Reload Safety Evaluation” (Reference 7). Specific differences in nuclear steam supply system (NSSS), reactor protection system (RPS), and fuel features between KPS and other Dominion Westinghouse units are capable of being reflected via modeling inputs in VEP-FRD-42 analytical methods, without changing the methodology. Implementation of this TR at KPS will be done in a manner consistent with the conditions and limitations identified in DOM-NAF-5. The staff finds the reload nuclear design methodology applicable to KPS as detailed in DOM-NAF-5.

3.2 Relaxed Power Distribution Control (RPDC) Methodology

The RPDC methodology, VEP-NE-1 (Reference 8), is a Dominion method for axial power distribution control with a variable axial flux difference (delta-I) band that provides an increasing delta-I band with decreasing power in order to maintain approximately constant analysis margin at all power levels. RPDC provides several operational benefits, such as increased ability to return to power after a trip, reduced control rod motion to compensate for delta-I band restrictions, and reduced reactor coolant system (RCS) boration and dilution requirements.

The RPDC analysis determines acceptable delta-I bands to maintain design bases margin. The process consists of: the generation of power shapes that bound the delta-I range; the selection of delta-I bands such that all bands satisfy the core operating limits report (COLR) height dependent hot channel factor, FQ(Z), limit with verification that the proposed delta-I bands satisfy LOCA FQ and loss of flow accident (LOFA) thermal-hydraulic evaluations; the examination of limiting Condition II events; the verification that over-power delta-temperature (OPΔT) and over-temperature delta-T (OTΔT) limits are conservative; and N(Z) functions are formulated to support the implementation of FQ TSs surveillance.

Similarity between the Dominion RPDC and Westinghouse, Combustion Engineering (CE), and Exxon-relaxed axial power distribution control methodologies is noted in DOM-NAF-5 and the VEP-NE-1 SER. DOM-NAF-5 also notes that the cooldown transient assumption differs between the RPDC methodology (20 °F) and the Westinghouse relaxed axial offset (RAOC) methodology currently used for KPS (30 °F); the larger value will be used unless a KPS-specific analysis demonstrates that a plant trip will occur before a 30 °F cooldown.

The VEP-NE-1 SER states that the RPDC approach is an acceptable methodology for use with reload cores similar to those of Surry Power Station (SPS) and North Anna Power Station (NAPS) because: approved methodologies are used for the analyses supporting RPDC; justification of uncertainties is provided; and the impact of cycle specific variations on the delta-I power domain, OPΔT and OTΔT trip setpoints, and other safety analyses are evaluated on a reload basis. In light of the similarity of the NSSS, RPS, and fuel design at KPS, SPS, and NAPS, the NRC staff finds that the RPDC approach is an acceptable methodology for use at KPS as documented in DOM-NAF-5.

3.3 Studsvik Core Management System Reactor Physics Methods

The Studsvik CMS reactor physics code package, detailed in TR DOM-NAF-1, "Qualification of the Studsvik Core Management System Reactor Physics Methods for Application to North Anna and Surry Power Stations," (Reference 9), consists of CASMO-4, SIMULATE-3, and CMS-LINK. CASMO-4 is a multi-group two-dimensional transport theory code for depletion and branch calculations for a single assembly that is used to generate the lattice physics parameters, including cross sections, nuclide concentrations, pin power distributions and other nuclear data, which are used as inputs to SIMULATE-3. SIMULATE-3 is a two-group, three-dimensional modified coarse-mesh nodal diffusion theory code with coupled thermal-hydraulic and Doppler feedback. CMS-LINK is a linking code that processes CASMO-4 card image files into a binary formatted nuclear data library for use by SIMULATE-3.

Dominion uses the Studsvik CMS package for startup physics testing, RPDC, and licensing applications, including core reload design, core operation, and key core parameters for reload safety analyses.

The Studsvik CMS benchmarking data provided in DOM-NAF-1 was based on the 15x15 and 17x17 fuel designs used at SPS and NAPS respectively, while KPS currently uses 14x14 fuel. In addition, DOM-NAF-1 SER limits the use of DOM-NAF-1, prohibiting its application to "significantly different or new fuel designs." Since this restriction is not clearly defined and in light of the absence of benchmarking data to 14x14 fuel, the KPS CMS models have been validated by comparison to benchmarks from both higher order Monte Carlo neutron transport calculations and reactor measurements from 10 cycles of operation spanning transitions in fuel enrichment, fuel density, spacer grid design, fuel vendor, core operating conditions and burnable poison design. These benchmarking results, as provided in Attachment A of DOM-NAF-5, are consistent with the staff approved methodology described in DOM-NAF-1. Thus, the NRC staff finds that the Studsvik CMS methodology as detailed in DOM-NAF-1 and DOM-NAF-5 is applicable to KPS.

3.4 Reactor System Transient Analyses using RETRAN

Dominion uses RETRAN to perform transient thermal-hydraulic analyses of the NSSS for best-estimate (e.g. training simulator validation) and licensing applications (e.g. reload core safety analysis), as detailed in TR VEP-FRD-41, "Vepco Reactor System Transient Analyses Using the RETRAN Computer Code" (Reference 10). RETRAN calculates general system parameters as a function of time and boundary conditions for input into more detailed calculations of departure from nucleate boiling (DNB) or other thermal and fuel performance margins.

The licensee performed transient analyses to confirm the adherence of reload core design limits to the bounds established by the reference analysis of record parameter values, as well as to verify that the core is acceptable from a safety operational point of view.

Transient analyses form an integral part of evaluations performed to verify the acceptability of a reload core design from the standpoints of safety and operational flexibility. The reload process consists of design initialization, design of the core loading pattern, and detailed characterization of the core loading pattern by the nuclear designer. The latter process determines the values of core physics related key analysis parameters. These key parameters are provided to the safety analyst, who uses them in conjunction with current plant operating configurations and limits to evaluate the impact of the core reload on plant safety.

The Dominion KPS RETRAN models have been validated by selecting representative transient events and comparing the results of the KPS RETRAN models to the vendor RETRAN model that was used to perform the current USAR analyses. This approach is similar to the one taken in VEP-FRD-41. The results of this analysis, as provided in Attachment B of DOM-NAF-5, show that the Dominion KPS RETRAN model compares favorably to the vendor RETRAN model for the selected transients, and the differences can be understood based on differences in noding, inputs, or other modeling assumptions. The NRC staff finds that the RETRAN methodology as detailed in DOM-NAF-5 and VEP-FRD-41 is applicable to KPS.

In performing this evaluation, it is necessary to ensure that those key parameters that influence accident response are maintained within the bounds or "limits" established by the parameter values used in the reference analysis (i.e. the currently applicable licensing calculation). The reference analysis (and the associated parameter limits) may be updated from time to time in support of a core reload or to evaluate the impact of some other plant parameter change.

In the case where a parameter is outside a previously defined limit, an evaluation of the impact of the change on the results for the appropriate transients is performed. This evaluation may be based on known sensitivities to changes in the various parameters in cases where a parameter change is small or the influence on the accident results is weak. For cases where larger parameter variations occur, or for parameters that have a strong influence on accident results, explicit reanalysis of the affected transients is required and performed. Past analytical experience has allowed the correlation of the various accidents with those parameters that have a significant impact on them.

If a reanalysis is performed, the results are compared to the appropriate analysis acceptance criteria. The reload evaluation process is complete if the acceptance criteria are met, and internal documentation of the reload evaluation is provided for the appropriate Dominion safety review. If the analysis acceptance criteria are not met, more detailed analyses and/or TS changes may be required to meet the acceptance criteria.

3.5 Statistical DNBR Evaluation Methodology

3.5.1 Analytical Methods

Topical Report DOM-NAF-5 details the events and analyses that will use the statistical DNBR evaluation methodology as well as those events that will use the deterministic models.

Topical Report VEP-NE-2, "Statistical DNBR Evaluation Methodology" (Reference 11), describes Dominion's methodology for statistically treating several of the important uncertainties in the DNBR analysis. The statistical DNBR evaluation methodology is used to determine a plant-specific and fuel-specific statistical DNBR limit. This limit DNBR combines the core heat flux (CHF) correlation uncertainty with DNBR sensitivities to uncertainties in key DNBR analysis input parameters. The statistical combination of some of these uncertainties permits a more realistic combination of the independent uncertainties and, thus, provides a more realistic evaluation of DNBR margin. The statistical DNBR evaluation methodology allows thermal-hydraulic evaluations to be performed using nominal operating conditions as opposed to deterministic initial conditions (nominal conditions plus evaluated uncertainty).

The statistical DNBR evaluation methodology is typically applied to all Condition I and II DNB events (except rod withdrawal from subcritical, RWSC), and to the LOFA analysis, the locked rotor accident and the single rod cluster control assembly withdrawal at power (SRWAP). The events modeled statistically (see Table 3.5.1 of Reference 1) are limited by the statistical design limits (SDLs) evaluated in the implementation of the statistical DNBR evaluation methodology for KPS, dated May 4, 2007 (Reference 4). In addition, there are events that will be evaluated with deterministic models. These events will be initiated from bounding operating conditions (nominal value), with appropriate uncertainty added to these nominal values. The events modeled deterministically are limited by the deterministic design limits (DDLs) stated in DOM-NAF-2 (Reference 12).

In its May 4, 2007, letter (Reference 4), Dominion submitted its KPS-specific statistical DNBR methodology analysis. The May 4, 2007, report supports the application of the NRC-approved TR stated above. In this plant-specific report, Dominion provided the technical basis and documentation necessary to evaluate the plant specific application of the VEP-NE-2-A methods to KPS. In its specific application analysis, Dominion used the VIPRE code with the Westinghouse WRB-1 CHF correlation for the thermal-hydraulic analysis of the Westinghouse 14x14 (422V+) fuel assemblies at KPS. The same report also provides documentation that the core safety limits and protection functions, such as the OTΔT, OPΔT, do not require revision as a consequence of this implementation.

3.5.2 Uncertainty Analysis

Consistent with VEP-NE-2-A, (Reference 11), various plant parameters were selected as the statistically treated parameters in the implementation analysis. The magnitudes and functional forms of the uncertainties for the statistically treated parameters were derived in a rigorous analysis of plant hardware, measurement and calibration procedures, and have been summarized in Table 3.2-1 of the May 4, 2007, submittal (Reference 4).

The uncertainties for core thermal power, vessel flow rate, pressurizer pressure and inlet temperature were quantified using all sensor, rack, and other components of a total uncertainty and combined in a manner consistent with their relative dependence or independence. Westinghouse quantified these uncertainties for Kewaunee's transition to Westinghouse's 14x14 (422V+) fuel. Total uncertainties were quantified at the 2 sigma level, corresponding to two-sided 95% probability.

The two-sided, 95/95 tolerance interval (95% probability, 95% confidence) for the measurement uncertainty of the nuclear enthalpy rise factor, $F_{\Delta H}$, is 3.5%. Conservatively, the measured $F_{\Delta H}$ uncertainty was defined as a normal distribution with a 4% tolerance interval for consistency with previous applications.

The magnitude and distribution of uncertainty on the enthalpy rise hot channel factor, $F_{\Delta H,}$ was quantified as a normal probability distribution with a magnitude of 3.0%. The statistical DNBR evaluation methodology (Reference 4) treats the $F_{\Delta H}$ uncertainty as a uniform probability distribution.

3.5.3 Verification of Nominal Set-points

Condition 1 of the NRC's SER for VEP-NE-2-A (Reference 11) requires that the nominal statepoints be shown to provide a bounding DNBR standard deviation for any set of conditions to which the methodology may be applied.

Consequently, in the May 4, 2007, submittal (Reference 4), the licensee provided analysis to demonstrate that S_{total} (the total DNBR standard deviation) as calculated in the TR is maximized for any conceivable set of conditions at which the core may approach the SDL. To this end, the licensee performed a regression analysis using as dependent variable the un-randomized DNBR standard deviations at each nominal statepoint (i.e. the raw MDNBR results obtained from a Monte Carlo simulation). The nominal statepoint pressures, inlet temperatures, powers and flow rates are used as the independent variable. The licensee stated that an evaluation of all the

data, linear fits, and regression coefficients indicated that there were no discernible trends in the database. Consequently, the licensee concluded that the total standard deviation had been maximized for any conceivable set of conditions at which the core may approach the SDL, and that the selected nominal statepoints provide a bounding standard deviation for any set of conditions to which the methodology may potentially be applied. The NRC staff finds the licensee's results and conclusion acceptable.

3.6 Reactor Core Thermal-Hydraulics using VIPRE-D Computer Code

The VIPRE (Versatile Internals and Components Program for Reactors) computer code is a reactor core thermal-hydraulics code developed by Battelle Pacific Northwest Laboratories. VIPRE is used to accurately calculate reactor coolant conditions to assure that the DNBR design limit is maintained.

The reactor core thermal-hydraulics code VIPRE-D, as described in TR DOM-NAF-2, is a Dominion-modified version of the VIPRE-01, MOD-02.1, which has been adapted to accommodate the various fuel designs used at Dominion nuclear power stations by incorporating vendor proprietary CHF correlations. The input and output has also been customized to incorporate it into the Dominion thermal hydraulic methodology.

VIPRE-D was approved by the NRC staff for pressurized-water reactor (PWR) licensing calculations up to the CHF using approved CHF correlations in accordance with the conditions and limitations listed in the SERs of DOM-NAF-2 and Electric Power Research Institute Report, NP-2511-CCM. In addition, VIPRE-D must be applied in a manner consistent with plant-specific and fuel-specific application conditions and limitations outlined in DOM-NAF-5. The NRC staff finds that the VIPRE-D thermal-hydraulics analysis methodology is applicable to KPS.

4.0 CONCLUSION

The NRC staff has reviewed Dominion's submittals and supporting documentation and finds the proposed use of Dominion nuclear core design and safety analysis methods at KPS to be acceptable. As such, TR DOM-NAF-5 is acceptable for use in licensing applications at KPS.

Based on the considerations discussed above, the NRC staff has concluded that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) such activities will be conducted in compliance with the Commission's regulations; and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 REFERENCES

1. Dominion Energy Kewaunee, Inc. (DEK) letter, Gerald T. Bischof to NRC, request for Approval of Topical Report DOM-NAF-5, “Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS),” August 16, 2006.

2. DEK letter, Gerald T. Bischof to NRC, “Attachment A to Dominion Energy Kewaunee, Inc., Kewaunee Power Station Request for Approval of Topical Report DOM-NAF-5, “Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS),” December 6, 2006.

3. DEK letter, Gerald T. Bischof to NRC, “Attachment B to Dominion Energy Kewaunee, Inc., Kewaunee Power Station Request for Approval of Topical Report DOM-NAF-5, “Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS),” April 16, 2007.

4. DEK letter, Gerald T. Bischof to NRC, “Dominion Energy Kewaunee, Inc., Kewaunee Power Station Request for Approval of Topical Report DOM-NAF-5, “Implementation of the Dominion Statistical DNBR Methodology with VIPRE-D/WRB-1 at Kewaunee Power Station (KPS),” May 4, 2007.

5. Dominion letter, Leslie N. Hartz to NRC, “Virginia Electric and Power Company Responses to NRC questions regarding Kewuanee request for approval of Topical Report DOM-NAF-5, “Application of Dominion Nuclear Core Design and Safety Analysis Methods to the Kewaunee Power Station (KPS),” June 12, 2007.

6. Dominion Topical Report VEP-FRD-42 Revision 2.0-A, “Reload Nuclear Design Methodology,” August 2003.

7. Westinghouse Topical Report WCAP-9272-P-A, 'Westinghouse Reload Safety Evaluation Methodology," (Proprietary), March 1978.

8. Dominion Topical Report VEP-NE-1, Rev. 0.1-A, "Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications," August 2003.

9. Dominion Topical Report DOM-NAF-I, Rev. 0.0-P-A, "Qualification of the Studsvik Core Management System Reactor Physics Methods for Application to North Anna and Surry Power Stations," June 2003.

10. Dominion Topical Report VEP-FRD-41, Rev. 0.1-A, "Vepco Reactor System Transient Analyses Using the RETRAN Computer Code," June 2004.

11. Dominion Topical Report VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," June 1987.

12. Dominion Topical Report DOM-NAF-2, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," September 2004.

Principal Contributor: A. Attard

Date: August 30, 2007

March 17, 2003

Mr. Thomas Coutu
Site Vice President
Kewaunee Nuclear Power Plant
Nuclear Management Company, LLC
N490 State Highway 42
Kewaunee, WI 54216

SUBJECT: KEWAUNEE NUCLEAR POWER PLANT - ISSUANCE OF AMENDMENT REGARDING IMPLEMENTATION OF ALTERNATE SOURCE TERM (TAC NO. MB4596)

Dear Mr. Coutu:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 166 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant (KNPP). This amendment revises the current radiological consequence analyses for the KNPP design-basis accidents in response to your application dated March 19, 2002, supplemented by letters dated September 13 and October 21, 2002.

The amendment revises the current radiological consequence analyses for the KNPP design-basis accidents to implement the alternate source term (AST) as described in Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors” and pursuant to 10 CFR 50.67, “Accident Source Term.” You did not request any changes to the current Technical Specifications for KNPP or any modifications to the plant design at this time.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

/RA/

John G. Lamb, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-305

Enclosures:
1. Amendment No. 166 to License No. DPR-43
2. Safety Evaluation

cc w/encls: See next page

March 17, 2003

Mr. Thomas Coutu
Site Vice President
Kewaunee Nuclear Power Plant
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cc w/encls: See next page

Distribution w/encls:

PUBLIC	WBeckner, TSB	PD 3-1 Reading	ACRS
OGC	GGrant, RIII	GHill (2)	

+See previous concurrence
*See memo F. Reinhart to L. Raghavan, dated 12/04/02

ADAMS Accession Number: ML030210062

OFFICE	PM:PD3-1	LA:PD3-1	SC:SPSB	SC:SPLB	OGC	SC:PD3-1
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DATE	03/11/03	03/13/03	12/04/02	03/12/03	3/10/03	03/12/03

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NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-305

KEWAUNEE NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 166
License No. DPR-43

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Nuclear Management Company, LLC (NMC or the licensee), dated March 19, 2002, supplemented by letters dated September 13 and October 21, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, by Amendment No. 166 Facility Operating License No. DPR-43 is hereby amended to revise the current radiological consequence analyses for the KNPP design-basis accidents to implement the alternate source term (AST) as described in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors" and pursuant to 10 CFR 50.67, "Accident Source Term" as set forth in the license amendment application dated March 19, 2002, supplemented by letters dated September 13 and October 21, 2002, and evaluated in the associated safety evaluation by the Commission's Office of Nuclear Reactor Regulation.

3. This license amendment is effective as of the date of its issuance, and is to be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Date of Issuance: March 17, 2003

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 166 TO FACILITY OPERATING LICENSE NO. DPR-43

NUCLEAR MANAGEMENT COMPANY, LLC

KEWAUNEE NUCLEAR POWER PLANT

DOCKET NO. 50-305

1.0 INTRODUCTION

By letter dated March 19, 2002, as supplemented by letters dated September 13 and October 21, 2002, Nuclear Management Company, LLC (NMC or the licensee), requested revisions to the current radiological consequence analyses for the Kewaunee Nuclear Power Plant (KNPP) design-basis accidents (DBAs). The licensee proposed to implement the alternative source term (AST) in this revision as described in Regulatory Guide (RG) 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors” and pursuant to 10 CFR 50.67, “Accident Source Term.” The current Kewaunee AST was developed using Technical Information Document (TID)-14844, “Calculation of Distance Factors for Power and Test Reactor sites.” The licensee has not requested any changes to the current KNPP Technical Specifications or any modifications to the plant design at this time.

The supplemental letters contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original *Federal Register* notice.

The proposed revisions would revise the current KNPP radiological consequence analyses for the following eight (8) DBAs described in Section 14 of the KNPP Updated Safety Analysis Report (USAR):

1. Loss-of-Coolant Accident (LOCA)
2. Steam Generator Tube Rupture (SGTR) Accident
3. Main Steamline Break (MSLB) Accident
4. Fuel Handling Accident (FHA)
5. Locked Rotor Accident (LRA)
6. Rod Ejection Accident (REA)
7. Gas Decay Tank Rupture (GDTR)
8. Volume Control Tank Rupture (VCTR)

ENCLOSURE

2.0 REGULATORY EVALUATION

Since the publication of TID-14844, significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents. The current radiological consequence analyses for the DBAs for KNPP are based upon the TID-14844 Accident Source Term. In 1995, the Nuclear Regulatory Commission (NRC) staff published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." NUREG-1465 provides estimates of the AST that were more physically based and that could be applied to the design of future light-water power reactors. NUREG-1465 presents a representative AST for a boiling-water reactor and for a pressurized-water reactor. These source terms are characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release to the containment.

The NRC staff considered the applicability of the revised source terms in NUREG-1465 to operating reactors and determined that the current analytical approach based on the TID-14844 source term would continue to be adequate to protect public health and safety. Operating reactors licensed under that approach would not be required to re-analyze accidents using the revised source terms. The NRC staff also determined that some licensees might wish to use an AST in their analyses to support cost-beneficial licensing actions. The NRC staff, therefore, initiated several actions to provide a regulatory basis for operating reactors to use an AST in design-basis radiological consequence analyses. These initiatives resulted in the development and issuance of 10 CFR 50.67 and RG 1.183. As part of the implementation of the AST, the total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67 replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19 as follows:

	10 CFR 50.67 GDC 19	10 CFR 100.11 GDC 19
Exclusion Area Boundary and Low Population Zone	25 rem TEDE	300 rem thyroid and 5 rem whole body
Control Room	5 rem TEDE	5 rem whole body, or its equivalent to any part of the body

A holder of an operating license issued prior to January 10, 1997, or a holder of a renewed license under 10 CFR Part 54 whose initial operating license was issued prior to January 10, 1997, is allowed by 10 CFR 50.67 to voluntarily revise its current AST used in design-basis radiological consequence analyses for a license amendment under 10 CFR 50.90. In this license amendment, the licensee requested a full-scope implementation of the AST, as described in RG 1.183 pursuant to 10 CFR 50.67 for changing the KNPP design-basis radiological consequence analyses for the DBAs. In general, information provided by RG 1.183 is reflected in Chapter 15.0.1 of the Standard Review Plan (SRP), “Radiological Consequence Analyses Using Alternative Source Term.”

Other relevant regulatory requirements applicable to this license amendment are (1) GDC 19, “Control Room” of Appendix A to 10 CFR Part 50, and NUREG-0737 III.D.3.4 as it relates to maintaining the control room in a safe, habitable condition under accident conditions by providing adequate protection against radiation, and toxic gases.

3.0 TECHNICAL EVALUATION

The licensee re-analyzed and submitted the radiological consequence analyses for 8 affected DBAs. The NRC staff requested in its Request for Additional Information (RAI), dated July 3, 2002, that the licensee provide the detailed radiological dose calculations complete with copies of the inputs prepared and outputs obtained from the computer code used in the dose calculations. In a response to the request, the licensee stated that the licensee’s contractor (Westinghouse) does not provide copies of proprietary dose calculation notes for NRC review, and instead, proposed to bring the dose calculation notes to the Westinghouse office in Rockville, MD for NRC review. The NRC staff accepted the licensee’s proposal and reviewed the following Westinghouse proprietary dose calculation notes provided at the Westinghouse office in Rockville, MD. The NRC staff confirmed that the methods used in these notes for dose calculations are consistent with the guidelines provided in Regulatory Guide 1.183 except as noted in this safety evaluation. These documents were prepared by Westinghouse in support of the licensee’s proposed revision to the KNPP design-basis radiological consequence analyses:

1. CN-CRA-99-01, Revision 1, “Kewaunee Steam Releases for Radiological Dose Calculations.”
2. CN-CRA-99-036, Revision 2, “Kewaunee SGTR Offsite Radiation Dose Analysis for Replacement Steam Generator Program.”
3. CN-CRA-00-69, Revision 0, “Kewaunee Main Steamline Break Doses using AST Methodology.”
4. CN-CRA-99-40, Revision 0, “Kewaunee Determination of Iodine Spray Removal Coefficients and DF Limits.”
5. CN-CRA-00-56, Revision 0, “Kewaunee AST FHA Doses.”
6. CN-CRA-99-29, Revision 1, “Kewaunee SGTR T&H Analysis for Replacement Steam Generator Program.”
7. CN-CRA-99-28, Revision 0, “Kewaunee Definition of Iodine Spike Rate and Duration.”
8. CN-CRA-99-46, Revision 2, “Kewaunee GDTR and VCTR Radiation Dose Analyses for Replacement Steam Generator Program.”
9. CN-CRA-00-2, Revision 1, “Kewaunee AST LOCA Doses.”
10. CN-CRA-00-68, Revision 0, “Kewaunee AST RCA Doses
11. CN-CRA-00-70, Revision 2, “Kewaunee AST REA Doses.”

The NRC staff considers the implementation of the AST to be a significant change to the KNPP design-basis. In order to accept the use of an AST and the TEDE dose criteria, the NRC staff must make a current finding of compliance with regulations applicable to this request. However, the NRC staff accepted prior KNPP design-basis assumptions or parameters that are unrelated to the use of the AST, or are unaffected by the AST and they were allowed to be continued as the licensee’s design-basis. The radiological consequence analyses are based on a reactor core power of 1650 MWt increased to 1683 MWt to cover uncertainty. The licensee conservatively increased the reactor core fission product inventory and the reactor coolant fission product concentrations by an additional 10 percent to allow future KNPP power uprating.

3.1 Loss-of-Coolant Accident

The current radiological consequence analysis for the postulated LOCA is based on the TID-14844 source term and it is provided in the KNPP USAR Section 14.3. To demonstrate that the engineered safety features (ESFs) systems designed to mitigate the radiological consequences following the postulated LOCA at Kewaunee will remain adequate after implementation of the AST, the licensee re-analyzed the offsite and control room radiological consequences of the postulated LOCA.

The licensee provided the results of its offsite and control room dose calculations and the major assumptions and parameters used in its dose calculations. As documented in its submittals, the licensee has determined that after implementation of the AST, the existing ESF systems at KNPP will provide assurance that the total radiological consequences of the postulated LOCA at the exclusion area boundary (EAB), in the low population zone (LPZ), and in the control room will meet the acceptable radiation dose criteria specified in 10 CFR 50.67 (b)(2). As part of the implementation of the AST, the TEDE acceptance criterion of 10 CFR 50.67 (b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and whole body dose guideline of GDC 19.

The NRC staff has reviewed the licensee's analyses and has performed an independent confirmatory radiological consequence dose calculation for the following three potential fission product release pathways after the postulated LOCA:

(1) containment leakage pathway,
(2) leakage pathway from ESF systems outside containment, and
(3) emergency core cooling system (ECCS) Recirculation back-leakage pathway to the Refueling Water Storage Tank (RWST).

3.1.1 Containment Leakage Pathway

The current KNPP containment design-basis leak rate is 0.5 weight percent per day. For the radiological consequence analysis, this rate is followed by 0.25 percent per day after 24 hours following a LOCA for the entire duration of the accident (30 days) consistent with the guidance provided in RG 1.183. Immediately following the accident, the shield building pressure increases due to heat transferred from the containment shell. Operation of one of two shield building ventilation fans will establish a negative pressure within the shield building. For the first 10 minutes into the postulated LOCA, it is assumed that 90 percent of the containment leakage is released directly to the environment without holdup or filtration. The licensee used this 10 minutes in its dose calculation instead of 6 minutes as described in the current KNPP USAR. The NRC staff finds this change to be acceptable since the 10 minute is more conservative. The remaining 10 percent of the containment leakage is assumed to enter the auxiliary building where it is filtered by the auxiliary building special ventilation system prior to release to the environment. These leakage assumptions are specified in the KNPP Technical Specification (TS) Section 6.20, "Containment Leakage Rate Testing Program," and described in more detail in Section 14.3.5 of the KNPP USAR.

Following achievement of a vacuum in the shield building at 10 minutes into the accident, only 1 percent of the containment leakage is assumed to be released directly to the environment without holdup or filtration, 10 percent continues to enter the auxiliary building where it is still

filtered by the auxiliary building special ventilation system (ABSVS) prior to release to the environment, and the remaining 89 percent is assumed to enter the shield building where it is filtered by the shield building ventilation system (SBVS) prior to release to the environment. These leakage assumptions are also specified in the KNPP TS Section 6.20, “Containment Leakage Rate Testing Program,” and described in more detail in Section 14.3.5 of the Kewaunee USAR.

Three time periods are associated with the shield building ventilation operation. During the first period (0 to 10 minutes), the SBVS starts and draws a vacuum in the shield building. During this period, the licensee assumed no credit for the shield building. The second period (10 to 30 minutes), the SBVS dampers will modulate to maintain a vacuum. During this period, the licensee assumed that the containment leakage (89 percent) will be processed by the SBVS filters prior to release to the environment. No credit is taken for recirculation. The final period (greater than 30 minutes) consists of stable system operation with a combination of recirculation and discharge to the environment to maintain the vacuum in the shield building. During this final period, the licensee has taken fission product removal credit by the SBVS filters for recirculation as well as release to the environment.

The fission products in the containment atmosphere following the postulated LOCA is mitigated by natural deposition processes and by the containment spray system (CSS). The licensee assumed a radioactive aerosol removal rate of 0.1 per hour in the containment atmosphere. This removal credit is taken after the CSS operation is terminated. The NRC staff finds 0.1 per hour aerosol removal rate to be reasonable (within the 85 percent of the uncertainty distribution) based on theits study published in NUREG/CR-6189, “A Simplified Model of Aerosol Removal by Natural Processes in Reactor containment,” and is therefore, acceptable.

The KNPP CSS is an ESF system and is designed to provide reactor building cooling and fission product removal in the containment following the postulated LOCA. The CSS consists of two spray pumps and they are automatically started by the coincidence of three sets of one out of two high-high containment pressure signals. The licensee assumed that one train of the CSS will operate following the postulated LOCA taking suction initially from the RWST with no startup delay until the water in the RWST reaches a pre-set low level in 0.91 hours after the accident. The spray pump suction is then transferred manually to the containment sump and the spray water is re-circulated. The licensee assumed fission product removal by the CSS during only initial spray operation and conservatively assumed no fission product removal during recirculation phase. The CSS is assumed to terminate at 0.91 hours.

The licensee used the models and guidance provided in RG 1.183 and SRP 6.5.2 , “Containment Spray as a Fission Product Cleanup System” to determine the removal rates by the CSS for iodine in elemental form and fission products in particulate form. The NRC staff finds that the removal rates calculated by the licensee are acceptable. The major parameters and assumptions used by the licensee including the spray removal rates are listed in Table 2. The containment leakage pathway contributed the most radiological consequence dose (greater than 97 percent) at the EAB, in the LPZ, and control room (see Table 1).

3.1.2 Post-LOCA Leakage Pathway From Engineered Safety Features Outside Containment

Any leakage water from ESF components located outside the containment releases fission products during the recirculating phase of long-term core cooling following a postulated LOCA. The leakage from ESF components occurs in the auxiliary building. The licensee conservatively assumed the leakage into the auxiliary building to start from the start of the accident. The leakage rate is assumed to be 6 gallons per hour. The licensee assumed that 1 percent of the total iodine activity in the leaked fluid becomes airborne and is released through the ABSVS to the environment during the entire period of the accident (30 days). The licensee further assumed that half of the iodine activity that becomes airborne from leak sources in the auxiliary building.

The iodine partition factor of 1 percent is a departure from the guidance provided in RG 1.183 which states that an iodine partition factor of 10 percent should be assumed unless a smaller amount can be justified based on the actual sump water pH history and area ventilation rate. The KNPP USAR Section 6.2.5, “Effects of Leakage from Residual Heat Removal System,” identifies that the temperature of the containment recirculation water is below 212 °F when ECCS recirculation begins and that any leakage liquid is therefore assumed to be sub-cooled and to remain liquid retaining iodine contained in any leakage water in the liquid.

The licensee stated that 1 percent iodine partition factor and 50 percent iodine plateout assumptions were contained in the original KNPP USAR Section 14.3.7 (original page 14.3-68) which were approved by the Atomic Energy Commission. Therefore, the licensee considers that these assumptions are parts of Kewaunee original license and that they are still current licensing bases. Furthermore, the NRC staff believes that this assumption is unrelated to the use of the AST. Therefore, the NRC staff finds that 1 percent iodine partition factor and 50 percent iodine plateout assumptions are acceptable. The radiological consequence contribution from this pathway is insignificant (less than 1 percent) at the EAB and to the control room operator (see Table 1).

3.1.3 Emergency Core Cooling System Back-Leakage Pathway to Refueling Water Storage Tank

Following a postulated LOCA, the suction water source for the ECCS is switched from the RWST to the containment sump. The leakage path back to the RWST is through the suction lines of the safety injection, internal containment spray, and residual heat removal pumps. The licensee conservatively assumed the leakage into the RWST to start from the start of the accident and a back-flow leakage of 3 gallons per minute (gpm). The licensee stated in their response dated August 23, 2002, to the NRC staff’s RAI that the back-leakage to the RWST is ensured not to exceed 3 gpm through a combination of visual inspections and hydraulic tests in accordance with KNPP Surveillance Procedure.

The licensee also assumed 1 percent iodine partition in the RWST leakage water as it did for the ECCS leakage to the auxiliary building. The licensee stated that in addition to this leakage water being below 212 °F, it is leaked into RWST water that will be significantly below 212 °F. The leakage water enters the RWST at the bottom of the tank. The NRC staff finds that the leakage rate and iodine partition factor assumed by the licensee are acceptable. The radiological consequence contribution from this pathway is less significant (less than 2 percent)

at the EAB for the postulated LOCA (see Table 1).

3.1.4 Radiological Consequence of Loss-of-Coolant Accident

The licensee reevaluated the radiological consequence resulting from the postulated LOCA using the AST and concluded that the radiological consequences at the EAB, LPZ, and in the control room are within the dose criteria specified in 10 CFR 50.67. The NRC staff has reviewed the AST implementation proposed by the licensee. In performing this review, the NRC staff relied upon information provided by the licensee, NRC staff experience in performing similar reviews, and where deemed necessary, on NRC staff's confirmatory calculations.

To verify the licensee's radiological consequence analyses, the NRC staff performed a confirmatory radiological consequence dose calculation. The results were also within the dose criteria specified in 10 CFR 50.67. Although, the NRC staff performed its independent radiological consequence dose calculations, as a means of confirming the licensee's results, the NRC staff's acceptance is based on the licensee's analyses. The results of the licensee's radiological consequence dose calculations are provided in Table 1 and the major parameters and assumptions used by the licensee and the NRC staff are listed in Tables 2 through 9. The radiological consequences at the EAB, and the LPZ, and in the control room, as calculated by the licensee and by the NRC staff, are all within the dose criteria specified in 10 CFR 50.67. Therefore, the NRC staff concludes that the proposed AST implementation revising the current design-basis radiological consequence analysis for the postulated LOCA is acceptable.

The bases for the NRC staff's acceptance are that the radiological consequences analyzed by the licensee at EAB, LPZ, and in the control room are within the dose criteria specified in 10 CFR 50.67.

3.2 Fuel Handling Accident

The current radiological consequence analysis for the postulated design-basis FHA is based on the accident source term described in TID-14844 and it is provided in KNPP USAR Section 14.2.1. The licensee re-evaluated the radiological consequences of a postulated FHA in the containment with no credit taken for containment isolation implementing the AST. Since the assumptions and parameters used for a FHA inside containment are identical to those for a FHA in the auxiliary building, the resulting radiological consequences are the same regardless of the location of the accident.

The licensee concluded in the submittals that the radiological consequences resulting from the postulated FHA in the containment with no credit taken for containment isolation are within the dose acceptance criteria specified in SRP 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," and GDC 19.

The licensee reached this conclusion as a result of:

(1) using the guidance provided in Appendix B to RG 1.183, "Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident,"

(2) taking no credit for containment isolation,

(3) taking no credit for removal of fission products by the spent fuel pool ventilation system in the auxiliary building,

(4) using an overall decontamination factor of 200 for iodine in elemental and particulate forms in the spent fuel pool water with minimum water depth of 23 feet consistent with the guidelines provided in RG 1.183,

(5) releasing all fission products within 2 hours using an exponential release model with higher release in the initial period,

(6) assuming all fuel rods in one fuel assembly with an axial power peaking factor of 1.7 are damaged to the extent that the entire gap activity inventory of the damaged fuel rods is released to the surrounding water,

(7) using a fission product decay period of 100 hours (time period from the reactor shutdown to the first fuel movement).

The NRC staff reviewed the licensee's methods, parameters, and assumptions used in its radiological dose consequence analyses and finds that they are consistent with the conservative guidance provided in RG 1.183. To verify the licensee's radiological consequence assessments, the NRC staff performed confirmatory radiological consequence dose calculations for the postulated FHA. The radiological consequences calculated by the NRC staff are well within the dose criterion specified in GDC 19 (5 rem TEDE in the control room), and meet the dose acceptance criterion specified in the SRP 15.0.1 (6.3 rem TEDE at the EAB).

Even though the NRC staff performed its confirmatory dose calculations, the NRC staff's acceptance is based on the licensee's analyses. The results of the licensee's radiological consequence calculations are provided in Table 1 and the major parameters and assumptions used by the licensee and acceptable to the NRC staff are listed in Table 3. The radiological consequences at the EAB, and the LPZ, and in the control room as calculated by the licensee are all well within the dose criterion specified in GDC 19 and meet the dose acceptance criterion specified in the SRP 15.0.1. Therefore, the NRC staff concludes that the proposed AST implementation revising the current design-basis radiological consequence analysis for the postulated FHA is acceptable.

3.3 <u>Steam Generator Tube Rupture Accident</u>

This DBA postulates a rupture in a tube in one of the two steam generators resulting in the transfer of reactor coolant water to the ruptured steam generator. The primary-to-secondary flow through the ruptured tube (break flow) following a SGTR results in a depressurization of the reactor coolant system (RCS), a reactor trip, and actuation of safety injection. After safety injection actuates, it is assumed that the RCS pressure will stabilize at a value at which the safety injection and break flows are equal. The break flow is assumed to continue until plant operators have taken action to reduce RCS pressure. When RCS pressure is less than the steam generator (SG) pressure, the pressure differential and the flow direction reverses, terminating the break flow. The licensee assumed this occurs within 30 minutes from safety

injection actuation.

At 8 hours after the accident, the licensee assumed the residual heat removal system (RHR) will be placed into service for heat removal and there will be no further steam release to the environment from the secondary system. Break flows, steam releases, and feedwater flows were determined using thermal-hydraulic analyses to bound the operating conditions. The analysis assumed that the noble gases entrained in the break flow are released to the environment without holdup or decontamination in the SGs. For the ruptured SG, the analysis assumed that part of the break flow will immediately flash to steam and the entrained gases be released to the environment with no holdup in the SGs. The portion of the break flow that does not flash is assumed to mix with the bulk water in the SGs and be released at the steaming rate of the SGs. The iodine release rates are reduced to account for partitioning between the liquid and vapor phases. The licensee provided two partitioning factors at pre-reactor trip and post-reactor trip. While the flash fraction would be greater before the trip, the associated releases would be via the main condenser which would afford iodine mitigation.

The licensee assumed the initial iodine inventory in the RCS and SG to be at the maximum concentrations permitted by TSs. The initial noble gas inventory in the RCS is based on fuel damage equivalent to 1.0 percent failed fuel. Two iodine spiking cases are considered. The first assumes that an iodine spike occurred just before the SGTR and that the RCS iodine inventory is at 60 µCi/gm dose equivalent I-131. The second case assumes the event initiates an iodine spike. In this case, iodine is released from the fuel to the RCS at a rate of 500 times the normal iodine appearance rate. This multiplier is more conservative than the 335 times multiplier specified by RG 1.183, and therefore, it is acceptable.

The licensee determined the iodine appearance rates assuming a letdown system flow rate of 88 gpm, radioactive decay, a primary system leakage of 12 gpm, and complete removal of iodine by the letdown system demineralizer (most conservative for the purpose of determining the iodine appearance rates). The NRC staff has verified the iodine appearance rates provided by its own calculations and finds that they are within the guidelines provided within Regulatory Guide 1.183. The licensee stated that the iodine release rate is such that the iodine inventory of the fuel rod gap will be depleted by four hours. In response to NRC staff questions, the licensee provided additional information on this assumption in their August 23, 2002, letter. The licensee assumed that the spike duration is a function of the accident-induced iodine appearance rate (148.6 Ci/min for iodine-131) and the iodine available for release from the fuel gap of fuel rods with defects. This evaluation yields a release duration of 3.4 hours. The NRC staff finds the licensee's justification of the 4-hour spike duration acceptable.

The licensee re-evaluated the radiological consequence resulting from the postulated SGTR accident and concluded that the radiological consequences at the EAB, LPZ and in the control room are within the dose acceptance criteria specified in SRP 15.0.1. The NRC staff reviewed the licensee's methods, parameters, and assumptions used in its radiological dose consequence analyses and finds that they are consistent with the conservative guidance provided in RG 1.183. To verify the licensee's radiological consequence assessments, the NRC staff performed confirmatory radiological consequence dose calculations for the postulated SGTR accident. The radiological consequences calculated by the NRC staff are within the dose criterion specified in GDC 19 (5 rem TEDE in the control room), and meet the dose acceptance criteria specified in the SRP 15.0.1. The radiological consequences calculated by the NRC staff are within 5 percent of those calculated by the licensee. The

results of the licensee's radiological consequence calculations are provided in Table 1 and the major parameters and assumptions used by the licensee and acceptable to the NRC staff are listed in Table 4.

3.4 Main Steamline Break Accident (MSLB)

This DBA postulates an unisolable failure in one of the two main steam lines at a location outside of containment, resulting in the release of steam from the affected steam line. The faulted SG will rapidly depressurize and release its entire liquid inventory and dissolved iodine through the faulted steam line to the environment. The rapid secondary depressurization causes a reactor power transient, resulting in a reactor trip. The released steam may be contaminated due to leakage of reactor coolant into the SGs via small tube leaks (i.e., primary-to-secondary leakage). The radiological consequences of a break outside containment will bound those results from a break inside containment. Thus, only the break outside containment is analyzed.

The licensee assumed that the faulted SG boils dry in two minutes and that primary-to-secondary leakage transfers 500 gallons per day (gpd), bounding the KNPP TS limit of 150 gpd per SG. This leakage mixes with the bulk SG water. Transferred noble gases are released without a holdup. Iodine is released to the environment at the steaming rate of the SGs with credit for iodine partitioning. The licensee assumed that the RHR system will be available for heat removal at 8 hours after the accident and that after 8 hours, there will be no further steam releases to the environment from the intact SG. The licensee further assumed that within 72 hours after the accident, the RCS has been cooled to below 212 °F, and there will be no further steam release from the faulted SG. The licensee's assumptions regarding the RCS inventory and iodine spiking are the same as those discussed above for the SGTR. No fuel damage is projected for the MSLB.

The licensee re-evaluated the radiological consequence resulting from the postulated MSLB accident and concluded that the radiological consequences at the EAB, LPZ and in the control room are within the dose criteria specified in 10 CFR 50.67. The NRC staff reviewed the licensee's methods, parameters, and assumptions used in its radiological dose consequence analyses and finds that they are consistent with the conservative guidance provided in RG 1.183. To verify the licensee's radiological consequence assessments, the NRC staff performed confirmatory radiological consequence dose calculations for the postulated MSLB accident. The radiological consequences calculated by the NRC staff are well within the dose criterion specified in GDC 19 (5 rem TEDE in the control room), and meet the dose acceptance criterion specified in the SRP 15.0.1. The radiological consequences calculated by the NRC staff are within 10 percent of those calculated by the licensee. The results of the licensee's radiological consequence calculations are provided in Table 1 and the major parameters and assumptions used by the licensee and acceptable to the NRC staff are listed in Table 5.

3.5 Locked Rotor Accident

For this DBA, a reactor coolant pump rotor is assumed to seize instantaneously causing a rapid reduction in the flow through the affected RCS loop. A reactor trip will occur, shutting down the reactor. The flow imbalance creates localized temperature and pressure changes in the core. If severe enough, these differences may lead to localized boiling and fuel damage. The radiological consequences are due to leakage of the contaminated reactor coolant to the SGs

and then releases from the SGs to the environment.

The licensee conservatively assumed that all of the fuel rods in the core are damaged and that all of its fuel gap activity is released to the reactor coolant system as a result of the primary coolant pump locked rotor accident. The NRC staff finds this assumption to be conservative based on the NRC staff's experience in performing similar reviews for other reactor plants. At 8 hours after the accident, the licensee assumed the RHR system will be placed into service for heat removal and there will be no further steam release to the environment from the secondary system. All other parameters and assumptions for fission product transport and release mechanisms are same as those discussed in the above Section 3.3, "SGTR Accident."

The NRC staff reviewed the licensee's methods, parameters, and assumptions used in its radiological consequence analyses and finds that they are consistent with the guidance provided in Regulatory Guide 1.183. To verify the licensee's radiological consequence assessments, the NRC staff performed confirmatory radiological consequence dose calculations for the postulated locked rotor accident.

The radiological consequences calculated by the licensee and by the NRC staff are within the dose criterion specified in GDC 19 (5 rem TEDE in the control room), and meet the dose acceptance criterion specified in the SRP 15.0.1. The results of the licensee's radiological consequence calculations are provided in Table 1 and the major parameters and assumptions used by the licensee and acceptable to the NRC staff are listed in Table 6.

3.6 Rod Ejection Accident

This DBA postulates the mechanical failure of a control rod drive mechanism pressure housing that results in the ejection of a rod cluster control assembly and drive shaft. Localized damage to fuel cladding and a limited amount of fuel melt are projected.

The release to the environment is assumed to occur through two pathways:

- Release of containment atmosphere (i.e., design leakage)
- Release of RCS inventory via primary-to-secondary leakage through SGs

The licensee assumed that 15 percent of the fuel rods suffer sufficient damage to result in the release of all of their gap inventory to the RCS or to the containment. The licensee further assumed that 10 percent of the core inventory of noble gases and iodine are in the fuel rod gap, consistent with the guideline provided in RG 1.183. A small fraction of the fuel in the failed rods is assumed to melt because of the accident. The licensee estimated this damage to be limited to 0.375 percent of the core. This is based on the assumption that 50 percent of the rods in departure from nuclear boiling undergo centerline melting and the melting is limited to the inner 10 percent occurring over 50 percent of the axial length of the affected rods. The NRC staff finds these assumptions to be conservative and therefore, acceptable.

The licensee assumed that 100 percent release of noble gases and 50 percent of iodine to the RCS, and 100 percent release of noble gases and 25 percent to the containment which are consistent with the guidelines provided in RG 1.183. The licensee conservatively assumed that a pre-accident iodine spike occurred just before the event such that the RCS specific activity has a concentration of 60 μCi/gm dose equivalent I-131.

For the containment leakage case, the iodine released is 95 percent cesium iodide, 4.85 percent elemental, and 0.15 percent organic. The licensee assumed the iodine species in the secondary system to be 97 percent elemental and 3 percent organic. The containment is projected to leak at its design leakage of 0.5 percent of its contents by weight per day for the first 24 hours and then at 0.25 percent for the remainder of the 30-day accident duration. These assumptions are also consistent with the guidelines provided in RG 1.183. For the containment leakage pathway, the licensee did not take credit for fission product removal by spray or aerosol deposition.

The NRC staff reviewed the licensee's methods, parameters, and assumptions used in its radiological consequence analyses and finds that they are consistent with the guidance provided in Regulatory Guide 1.183. To verify the licensee's radiological consequence assessments, the NRC staff performed confirmatory radiological consequence dose calculations for the postulated rod ejection accident.

The radiological consequences calculated by the licensee and by the NRC staff for the rod ejection accident are within the dose criterion specified in GDC 19 (5 rem TEDE in the control room), and meet the dose acceptance criteria specified in the SRP 15.0.1. The results of the licensee's radiological consequence calculations are provided in Table 1 and the major parameters and assumptions used by the licensee and acceptable to the NRC staff are listed in Table 7.

3.7 <u>Gas Decay Tank and Volume Control Tank Rupture Accidents</u>

The KNPP licensing basis includes analyses of the radiological consequences of a rupture of a gas decay tank (GDT) and a rupture of the volume control tank (VCT). The current SRP does not classify these accidents as a DBA. The GDTs are used to store processed radioactive gases removed from the reactor coolant to allow for radioactive decay before the controlled release to the environment. The VCT is a component in the plants' chemical and volume control systems that serves as a surge volume to balance differences in letdown and makeup flow rates while maintaining reactor coolant inventory. Part of the reactor coolant (known as letdown) is removed from the RCS, cooled, filtered, demineralized, and degassed.

The GDT rupture case assumes that the entire inventory of gases in the RCS based on continuous operation with 1.0 percent failed fuel is in the GDT and is released over a period of five minutes. The initial inventory of noble gases in the VCT is based on continuous operation with 1.0 percent failed fuel, without purging of the VCT. The initial inventory of iodine is based on continuous operation with an RCS specific inventory of 60 µCi/gm dose equivalent I-131, with a 90 percent removal by the letdown demineralizer. The initial inventory is assumed to be released over a period of five minutes. After the event starts, letdown flow to the VCT is assumed to continue at the maximum flow rate of 88 gpm for 5 minutes, after which letdown is isolated. Iodine in the letdown flow are reduced by 90 percent by the letdown demineralizer.

To verify the licensee's radiological consequence assessments, the NRC staff performed confirmatory radiological dose calculations for the postulated GDT and VCT rupture accidents.

The radiological consequences calculated by the licensee and by the NRC staff for the GDT rupture and VCT rupture are small fractions (less than 1 percent) of the dose criteria specified in 10 CFR 50.67 and therefore, the NRC staff finds them acceptable. The major parameters

and assumptions used by the licensee and acceptable to the NRC staff are listed in Table 8.

3.8 Control Room Habitability

The KNPP control room habitability systems include radiation shielding, two control room air conditioning units, redundant control room post-accident recirculation filter/fan units, control room isolation, radiation monitoring, and redundant outside air intakes. During normal operation, one air conditioning unit recirculates control room air at 13,450 cubic feet per minute (cfm). This air is mixed with 2,500 cfm of unfiltered outside makeup air.

Upon a safety injection signal and/or high radiation detection, the control room is automatically isolated, 100 percent of control room air is recirculated, and 2,500 cfm of the recirculation flow is processed through one of two control room post-accident recirculation filter/fan units.

Each control room post-accident recirculation filter/fan unit consists of, among other things, a pre-filter, high efficiency particulate air filter, and 2-inch deep charcoal adsorber.

In 1989, the licensee submitted an updated control room habitability evaluation report in response to the NRC concerns expressed in NUREG/CR-4960, “Control Room Habitability Survey of Licensed Commercial Nuclear Power Generating Stations (1988).” The NUREG/CR-4960 presented the results of a survey of control room habitability systems at 12 nuclear power plants. In its updated control room habitability evaluation report, the licensee documented actions taken to address the NRC concerns including unfiltered air inleakage into the control room.

The licensee stated in the report that they performed air flow measurements in the system using an electronic micro-manometer to determine the unfiltered air inleakage into the control room envelope and the system performance. The licensee did not performed an integrated control room unfiltered air inleakage test. Instead, the licensee estimated the unfiltered air inleakage to be 200 cfm based on the air flow measurements. The NRC staff accepted the licensee’s report and 200 cfm unfiltered air inleakage rate assumption in both KNPP License Amendment No. 88 issued in 1990 and in KNPP License Amendment No. 132 issued in 1997. The licensee considers the 200 cfm as the current KNPP licensing and design bases.

In its response dated September 13, 2002, to the NRC staff’s RAI, the licensee stated that the unfiltered inleakage was determined by using the measured leakage through the closed dampers (48 cfm), allowance for leakage through building elements (80 cfm), and air exchange based on door opening and closing (10 cfm). The leakage was adjusted based on the worse case leakage resulting from one of the redundant dampers failing to close. The licensee stated that this resulted in a total leakage of 200 cfm.

The NRC staff is currently working toward a resolution of generic issues related to control room habitability, with particular focus on the validity of the control room unfiltered air inleakage rates that are commonly assumed in licensee’s analyses of control room habitability. The NRC staff’s acceptance of the 200 cfm unfiltered air inleakage assumption in this license amendment does not preclude any future generic regulatory actions that may become applicable to KNPP.

This assessment may be used in subsequent amendments; however, any use of this assessment that involves a relaxation in requirements will require verification (in accordance

with the aforementioned resolution of the generic issues related to control room habitability) that the unfiltered inleakage rate is within limits assumed in the AST assessment.

The NRC staff has determined that there is reasonable assurance that the KNPP control will be habitable with up to 200 cfm unfiltered air inleakage during the postulated DBAs and this amendment may be approved before the resolution of control room generic issue. The NRC staff bases this determination on (1) the maximum allowable unfiltered air inleakage rate of up to 200 cfm meeting the relevant dose acceptance criteria specified in 10 CFR 50.67 and GDC 19, (2) conservative assumptions and parameters used in the radiological consequence analyses, and (3) the low probability of the postulated accidents, occurring during this interim period until the NRC staff resolves this generic issue, that could result in radioactivity releases sufficient to challenge the ability of control room operators to protect the health and safety of the public.

3.9 Technical Conclusion

The NRC staff has reviewed the licensee's analyses and performed confirmatory assessments of the radiological consequence of the postulated DBAs. The doses calculated by the licensee are listed in Table 1. The doses calculated by the licensee and by the NRC staff are all within relevant dose criteria specified in 10 CFR 50.67 and SRP 15.0.1. Therefore, the NRC staff concludes that the radiological consequences analyzed and submitted by the licensee are acceptable. The NRC staff further concludes that the implementation of the AST replacing the current KNPP TID-14844 source term for the KNPP DBAs identified in the licensee's submittal is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Wisconsin State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

This 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 staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding (67 FR 18646). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

6.0 CONCLUSION

The NRC staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: J. Lee

Date: March 17, 2003

Table 1
Radiological Consequences Expressed as TEDE
(rem)

Design-basis Accidents	EAB[1]	LPZ[2]	Control Room
LOCA			
Containment leakage pathway	1.26	0.221	4.35
ECCS leakage pathway	1.44E-3	1.16E-3	2.79E-2
RWST leakage pathway	4.33E-2	1.23E-2	5.66E-1
Total (rounded up)	1.31	0.24	4.95
Dose acceptance criteria[3]	25	25	5.0
Fuel handing accident	0.6	0.11	1.0
Dose acceptance criteria	6.3[4]	6.3[4]	5.0[3]
Steam generator tube rupture			
Pre-accident iodine spike	1.3	0.3	3.0
Dose acceptance criteria[3]	25	25	5.0
Accident-initiated iodine spike	0.8	0.2	1.0
Dose acceptance criteria	2.5[4]	2.5[4]	5.0[3]
Main steamline break			
Pre-accident iodine spike	0.05	0.02	0.7
Dose acceptance criteria[3]	25	25	5.0
Accident-initiated iodine spike	0.2	0.05	2.3
Dose acceptance criteria	2.5[4]	2.5[4]	5.0[3]
Locked rotor accident	1.7	0.3	4.3
Dose acceptance criteria	2.5[4]	2.5[4]	5.0[3]
Control rod ejection accident	0.52	0.11	1.9
Dose acceptance criteria	6.3[4]	6.3[4]	5.0[3]

[1] Exclusion area boundary
[2] Low population zone
[3] 10 CFR 50.67
[4] SRP 15.0.1

Table 2
Parameters and Assumptions Used in
Radiological Consequence Calculations
Loss-of-Coolant Accident

Parameter	Value
Reactor power	1851.3 MWt
Containment volume	1.32E+6 ft^3
Containment leak rates	
0 to 1 hour	0.5 percent per day
1 to 720 hours	0.25 percent per day
Aerosol removal rate (after 0.91 hours)	0.1 per hour
Iodine removal rates by spray (per hour)	
0 to 0.91 hours	
Elemental	20
Particulate	5
0.91 to 720 hours	
Elemental	0
Particulate	0
Containment sump volume	4.21E+4 ft^3
ECCS leak rate	
0 to 720 hours	6 gph
Iodine partition factor	1 percent
ECCS leak rate to RWST	
0 to 24 hours	3.0 gpm
24 to 720 hours	1.5 gpm
Iodine partition factor	1 percent
Control room	
Volume	1.276E+5 ft^3
Filtered makeup air flow	0
Filtered Recirculation air flow	2500 cfm
Unfiltered air inleakage rate	200 cfm
Filter efficiencies	
Aerosol	99 percent
Elemental iodine	90 percent
Organic iodine	90 percent

Table 3
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Fuel Handling Accident

Parameter	Value
Reactor power	1851.3 MWt
Radial peaking factor	1.7
Fission product decay period	100 hours
Number of fuel assembly damaged	1
Fuel pool water depth	23 ft
Fuel gap fission product inventory	
Noble gases excluding Kr-85	5 percent
Kr-85	10 percent
I-131	8 percent
Alkali metals	12 percent
Fuel pool decontamination factors	
Iodine	200
Noble gases	1
Duration of accident	2 hours

Table 4
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Steam Generator Tube Rupture Accident

Reactor coolant activity	
Accident-initiated iodine spike case	1.0 µCi/gm dose equivalent I-131
Pre-accident iodine spike case	60.0 µCi/gm dose equivalent .I-131
Primary-to-secondary break flow	
Pre-trip	1.68E+4 lbm
Post-trip	1.38E+5 lbm
Break flow flash fractions	
Pre-trip	19.71
Post-trip	14.76
Break flow duration	30 minutes
Steam generator mass	2.685E+5 lbm
Steam release from ruptured steam generator	8.5E+4 lbm (post-trip)
Steam release from intact steam generator	
0-2 hours	2.438E+5 lbm
2-8 hours	5.066E+5 lbm
Intact steam generator iodine activity	0.1 µCi/gm dose equivalent .I-131
Steam iodine partition coefficient	0.01
Intact steam generator tube leak rate	500 gallons per day
Iodine spiking factor	500
Loss of offsite power	At reactor trip
Reactor trip time	173.7 seconds

Table 5
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Main Steam Line Break Accident

Reactor coolant activity	
Accident-initiated iodine spike case	1.0 µCi/gm dose equivalent I-131
Pre-accident iodine spike case	60.0 µCi/gm dose equivalent .I-131
Steam generator mass	2.685E+5 lbm
Steam release from faulted steam generator	1.5E+5 lbm
Time to release initial mass in faulted steam generator	2 minutes
Steam release from intact steam generator	
0-2 hours	2.2E+5 lbm (+10 percent)
2-8 hours	3.93E+5 lbm (+10 percent)
Intact steam generator activity	0.1 µCi/gm dose equivalent .I-131
Steam partition coefficient	
Faulted steam generator	1
Intact steam generator	0.01
Steam generator tube leak rate	500 gallons per day per steam generator
Iodine spiking factor	500
Duration of iodine spike	4 hours

Table 6
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Locked Rotor Accident

Reactor coolant activity prior to accident	60.0 µCi/gm dose equivalent .I-131
Secondary coolant activity prior to accident	0.1µCi/gm dose equivalent .I-131
Fraction of fuel rods failed	100 percent
Gap fractions	Per Regulatory Guide 1.183
Steam release from steam generator	
0-2 hours	2.13E+5 lbm (+10 percent)
2-8 hours	4.18E+5 lbm (+10 percent)
Steam generator tube leak rate	500 gallons per day per steam generator
Iodine partition factor in steam generator	0.01

Table 7
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Rod Ejection Accident

Reactor coolant activity prior to accident	60.0 µCi/gm dose equivalent .I-131
Secondary coolant activity prior to accident	0.1µCi/gm dose equivalent .I-131
Fraction of fuel rods failed	15 percent
Gap fractions(percent)	
Iodine	10
Noble gas	10
Alkali metals	12
Fraction of fuel melted (percent)	0.375
Fraction activity released from melted fuel (percent)	
For containment leakage pathway	
Iodine	25
Noble gas	100
Alkali metals	100
For primary to secondary leakage pathway	
Iodine	50
Noble gas	100
Alkali metals	100
Steaming rates (lbm per second)	
0-200 seconds	800
222 -1800 seconds	100
After 1800 seconds	0
Steam generator tube leak rate	500 gallons per day per steam generator
Iodine partition factor in steam generator	0.01
Containment leak rates	
0 to 1 hour	0.5 percent per day
1 to 720 hours	0.25 percent per day
Aerosol removal rate in containment	0

Table 8
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Gas Decay Tank and Volume Control Tank Ruptures

Reactor coolant iodine activity prior to accident	60.0 µCi/gm dose equivalent .I-131
Reactor coolant noble gas activity prior to accident	1 percent fuel defect
Release duration (minutes)	
Gas decay tank	5
Volume control tank	5
Letdown flow rate to volume control tank	88 gpm
Time to isolate letdown flow	5 minutes
Letdown demineralizer decontamination factor	10

Table 9
Meteorological Data
used for
Radiological consequence Analyses

Exclusion Area Boundary [(1)]

Time (hr)	X/Q (sec/m^3)
0-2	2.232E-4

Low Population Zone Distance [(1)]

Time (hr)	X/Q (sec/m^3)
0 to 2	3.977E-5
2 to 24	4.100E-6
24 to 48	2.427E-6
48 to 720	4.473E-7

Control Room [(2)]

Time (hr)	X/Q (sec/m^3)
0 to 8	2.93E-3
8 to 24	1.73E-3
24 to 96	6.74E-4
96 to 720	1.93E-4

(1) Original licensing basis in Kewaunee USAR, Table 14.3-8
(2) Updated Control Room Habitability Evaluation Report, Table 5, based on “Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criteria 19,” K. G. Murphy and K. M. Campe, 13th Air Cleaning Conference, August 1974.

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February 27, 2004

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SUBJECT: KEWAUNEE NUCLEAR POWER PLANT - ISSUANCE OF AMENDMENT REGARDING STRETCH POWER UPRATE (TAC NO. MB9031)

Dear Mr. Coutu:

The Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 172 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant (KNPP). This amendment revises the Operating License and Technical Specifications (TSs) in response to your application dated May 22, 2003, as supplemented July 9, November 5, and December 15, 2003, and January 30, February 9, and February 20, 2004.

The amendment authorizes an increase in the licensed reactor core power level of 6.0 percent from 1673 megawatts thermal to 1772 megawatts thermal.

We would like to share with you our observations regarding the quality and completeness of your power uprate submittals. Several areas of your application lacked sufficient technical details and your supplements in response to our requests for additional information (RAIs) did not completely address the requested information. This resulted in iterations of RAIs and challenged our ability to conduct a comprehensive review consistent with our timeliness goals and your proposed schedule. We request that you take appropriate corrective actions to improve your performance in these areas which will enable us to achieve the NRC's effectiveness and efficiency performance goals.

A copy of the safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice.

Sincerely,

/RA/

John G. Lamb, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-305

Enclosures: 1. Amendment No. 172 to License No. DPR-43
2. Safety Evaluation

cc w/encls: See next page

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February 27, 2004

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Sincerely,
/RA/
John G. Lamb, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-305
Enclosures: 1. Amendment No. 172 to License No. DPR-43
2. Safety Evaluation
cc w/encls: See next page
<u>DISTRIBUTION</u>:

PUBLIC	SWeerakkody	PGill	EWeiss	MShuaibi
PDIII-1 Reading	OGC	DTrimble	ALund	LBMarsh
LRaghavan	ACRS	SCoffin	JUhle	ELeeds
JLamb	GHill(2)	KManoly	FReinhart	DSolario
THarris	PLouden, RIII	EMarinos	WRuland	

*Provided SE input by memo
**Previously Concurred

OFFICE	PDIII-1/PM	PDIII-1/LA	SPLB/SC	SPLB/SC	EMCB/SC	EMCB/SC	EMCB/SC	EEIB/SC	EEIB/SC
NAME	JLamb	THarris RB chgs only	SWeerakkody*	DSolario*	SCoffin*	TChan*	ALund*	EMarinos*	RJenkins*
DATE	02/27/04	02/25/04	02/12/04	2/27/04	11/05/03	02/20/04	12/15/03	02/06/04	02/05/04

OFFICE	IEHB/SC	EMEB/SC	SRXB/SC	SPSB/SC	OGC	PDIII-1/SC	PDIII/D	DLPM/D
NAME	DTrimble**	KManoly*	JUhle**	RDennig*	RHoefling**	LRaghavan	WRuland	ELeeds for LMarsh
DATE	02/13/04	02/13/04	02/19/04	02/02/04	02/18/04	02/27/04	02/27/04	02/27/04

ADAMS Accession No. ML040430633

Kewaunee Nuclear Power Plant

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NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-305

KEWAUNEE NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 172
License No. DPR-43

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Nuclear Management Company, LLC (NMC or the licensee), dated May 22, 2003, as supplemented July 9, November 5, and December 15, 2003, and January 30, February 9, and February 20, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-43 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 172, are hereby incorporated in the license. The licensee's shall operate the facility in accordance with the Technical Specifications.

In addition, the license is amended to revise paragraph 2.C.(1) to reflect the increase in the reactor core power level. Paragraph 2.C.(1) is hereby amended to read as follows:

> The NMC is authorized to operate the facility at steady-state reactor core power levels not in excess of 1772 megawatts (thermal).

3. This license amendment is effective as of the date of its issuance, and is to be implemented within 90 days of the date of issuance, unless as otherwise noted. In addition, the licensee shall:

   A. Revise all documents, as appropriate, for the stretch power uprate to address Framatome ANP fuel departure-from-nucleate boiling ratio (DNBR) design basis prior to implementation of the license amendment.

   B. Complete changes to the condensate storage tank level setpoints, and first-stage turbine pressure, as appropriate, prior to implementation of the license amendment.

   C. Complete piping and pipe support evaluations for service water and component cooling water prior to implementation of the license amendment.

   D. Update the Kewaunee Nuclear Power Plant Environmental Qualification Plan, as appropriate, to reflect power uprate evaluations prior to implementation of the license amendment.

   E. Revise plant procedures, as appropriate, to accommodate the stretch power uprate prior to implementation of the license amendment. Emergency, abnormal, and operating procedures that are entered due to a loss-of-normal feedwater event or have auxiliary feedwater (AFW) Technical Specification requirements shall be changed, as appropriate, prior to implementation of the license amendment.

   F. Review new Technical Specification requirements, revised procedures, and any control room changes due to the stretch power uprate to determine necessary changes to the operator training program and complete the required portion of the operator training prior to implementation of the license amendment.

   G. Update setpoint changes for reactor protection and control inputs, alarms, computer constants, and embedded values, consistent with operation at 1772 MWt prior to implementation of the license amendment. Power range nuclear instruments shall be recalibrated and checked based on a secondary heat balance prior to implementation of the license amendment.

H. Revise degraded voltage and thrust calculations for motor-operated valve operators outside containment which were reviewed for impact of uprated post accident temperatures, as required, prior to implementation of the license amendment.

I. Provide NRC with a status update and schedule for resolution of Generic Letter (GL) 96-06 water hammer issues by April 2, 2004.

J. Incorporate the increase in flow rate and velocities, as well as the changes in operating pressures and temperatures, into the Kewaunee Nuclear Power Plant Flow Accelerated Corrosion (FAC) Program as part of the power uprate implementation. The Kewaunee Nuclear Power Plant FAC program models shall be updated prior to the next program inspections scheduled for the next refueling outage.

K. Establish an inspection and monitoring program to monitor potential feedwater heater degradation at the stretch power uprate conditions. An inspection program shall be developed based on the baseline inspection results and using programs and processes in place at the Kewaunee Nuclear Power Plant. This shall be completed prior to the next refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by ELeeds for/

Ledyard B. Marsh, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Operating License and Technical Specifications

Date of Issuance: February 27, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 172

FACILITY OPERATING LICENSE NO. DPR-43

DOCKET NO. 50-305

Replace the following page of Facility Operating License No. DPR-43 with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
3	3

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
TS 1.0-4	TS 1.0-4
TS 2.1-1	TS 2.1-1
TS B 2.1-1	TS B 2.1-1
TS B 2.1-2	TS B 2.1-2
TS 2.3-2	TS 2.3-2
TS 2.3-3	TS 2.3-3
TS B 2.3-2	TS B 2.3-2
TS 3.3-4	TS 3.3-4
TS B 3.3-3	TS B 3.3-3
TS 3.4-2 through TS 3.4-4	TS 3.4-2 through TS 3.4-3
TS B 3.4-2 through TS B 3.4-4	TS B 3.4-2 through TS B 3.4-4
TS B 3.8-1	TS B 3.8-1
TS B 3.8-2	TS B 3.8-2
Table TS 3.5-1	Table TS 3.5-1
TS B 4.5-1	TS B 4.5-1
Table TS 4.1-2	Table TS 4.1-2

Kewaunee Nuclear Power Plant

Safety Evaluation for Amendment No. 172

Stretch Power Uprate

TABLE OF CONTENTS

Attachments: 1) List of Acronyms
2) Tables 1 through 10

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 172 TO FACILITY OPERATING LICENSE NO. DPR-43

NUCLEAR MANAGEMENT COMPANY, LLC

KEWAUNEE NUCLEAR POWER PLANT

DOCKET NO. 50-305

1.0 INTRODUCTION

By application dated May 22, 2003, as supplemented July 9, November 5, and December 15, 2003, and January 30, February 9, and February 20, 2004, the Nuclear Management Company, LLC (NMC or the licensee), requested an amendment to the Facility Operating License and the Technical Specifications (TSs) for the Kewaunee Nuclear Power Plant (KNPP). The proposed amendment would increase the licensed reactor core power level by 6.0 percent from 1673 megawatts thermal (MWt) to 1772 MWt. Based on its review of this application, the Nuclear Regulatory Commission (NRC) staff categorized the application as a stretch power uprate request based on the limited plant modifications required to achieve the requested power level. The plant modifications required to achieve the 6.0 percent stretch power uprate are (1) modification of the valve trim in the feedwater control valve, (2) replacement of the high-pressure turbine outer cylinder horizontal joint bolting to accommodate the higher loading conditions, and (3) replacement of the low-pressure turbine-to-jackshaft and low-pressure turbine-to-generator coupling bolts with higher strength material.

Specifically, the following are the proposed changes:

1. Paragraph 2.C.(1) of the operating license, DPR-43, would be revised to authorize operation at reactor core power levels not in excess of 1772 MWt.

2. TS 1.0.m, RATED POWER, would be revised to reflect the increase from 1673 MWt to 1772 MWt.

3. TS 2.1.c, regarding peak fuel centerline temperature, would be revised to increase the peak fuel centerline temperature from < 4700 °F to < 5080 °F. Additionally, the following text would be inserted at the end of TS 2.1.c, "decreasing by 58 °F per 10,000 MWD/MTU of burnup."

4. Table TS 3.5-1, "Engineered Safety Features Initiation Instrument Setting Limits," setting limit for 6, "High-High Steam Flow in a Steam Line Coincident with Safety Injection," would be revised from 4.5×10^6 pounds per hour (lb/hr) at 735 pounds per square inch gauge (psig) to 4.4×10^6 lb/hr at 735 psig.

5. TS 3.3.c.1.A.3 (iii), which allows both containment fancoil unit (CFCU) trains to be out of service for 72 hours provided both containment spray trains remain operable, would be deleted in its entirety and the subsequent item, (iv), renumbered as (iii).

6. TS 3.4.c.1, “Condensate Storage Tank,” would be revised to increase the minimum volume from 39,000 gallons to 41,500 gallons. “Usable volume” would be added to the specification for clarification.

7. TS 3.4.c.2, “Condensate Storage Tank,” would be revised to increase the minimum volume from 39,000 gallons to 41,500 gallons. “Usable volume” would be added to the specification for clarification.

8. TS 2.3.a.3.A, for $f(\Delta I)$, would be revised to change “An even function” to “A function.”

9. TS 2.3.a.3.A, for $f(\Delta I)$, would be revised to change $f(\Delta I)$ to $f_1(\Delta I)$. TS 2.3.a.3.B, for $f(\Delta I)$, would be revised to change $f(\Delta I)$ to $f_2(\Delta I)$.

10. Table TS 4.1-2, “Minimum Frequencies for Sampling Tests,” would be revised to change the units in the frequency column for sampling test 7, “Secondary Coolant, b. Iodine Concentration,” from 0.1 µCi/cc to 0.1 µCi/gram.

11. TS 3.4.b regarding the auxiliary feedwater (AFW) system would be revised to require three operable AFW trains prior to increasing reactor power above 1673 MWt. AFW trains are defined in current TS Basis Section 3.4.b, page TS B3.4-1. The original TS requirements within the section would be reordered to accommodate the new TS requirement as follows:

    - TS 3.4.b.2 would be renumbered as 3.4.b.4.
    - New TS 3.4.b.3 would be inserted stating the following: “The reactor power shall not be increased above 1673 MWt unless three trains of AFW are OPERABLE. If two of three AFW trains are inoperable, then within two hours, action shall be taken to reduce reactor power to $\leq$ 1673 MWt.”
    - TS 3.4.b.3 would be renumbered as 3.4.b.5 and the reference to TS 3.4.b.2 in the last sentence would be revised to reference TS 3.4.b.3 and TS 3.4.b.4.
    - TS 3.4.b.4 would be renumbered as 3.4.b.2.
    - TS 3.4.b.5 would be renumbered as 3.4.b.6 and reference to TS 3.4.b.2 would be replaced with TS 3.4.b.4.
    - TS 3.4.b.6 would be renumbered as 3.4.b.7.

The July 9, November 5, and December15, 2003, and January 30, February 9, and February 20, 2004, supplements provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff’s original proposed no significant hazards consideration determination as published in the *Federal Register* on June 10, 2003 (68 FR 34670).

2.0 BACKGROUND

The NRC's approval of the licensee's stretch power uprate request was contingent upon the NRC's approval of the following previous license amendments:

1. License Amendment No. 166, dated March 17, 2003 (alternate source term (AST) methodology for design-basis radiological accident analysis)
2. License Amendment No. 167, dated April 4, 2003 (fuel transition)
3. License Amendment No. 168, dated July 8, 2003 (1.4-percent measurement uncertainty recapture (MUR) power uprate)
4. License Amendment No. 169, dated September 29, 2003 (Generation of Thermal-Hydraulic Information for Containment (GOTHIC) version 7.0p2 (GOTHIC 7) computer code).

3.0 EVALUATION

The U.S. Atomic Energy Commission (AEC) issued a "Safety Evaluation of the Kewaunee Nuclear Power Plant" on July 24, 1972, as supplemented December 18, 1972, and May 10, 1973. The AEC performed a technical review of the KNPP against the General Design Criteria (GDC) in effect at the time and concluded that the KNPP design generally conforms to the GDC.

In several places in this safety evaluation (SE), the NRC staff refers to NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," as guidance used during the review. The NRC staff notes that the SRP was used solely for general technical guidance. The licensee's May 22, 2003, application, supplemented July 9, November 5, and December 15, 2003, was reviewed for compliance with the KNPP licensing basis, not NUREG-0800.

3.1 Instrumentation and Controls

3.1.1 Regulatory Evaluation

The NRC previously approved a 1.4-percent MUR power uprate for KNPP by License Amendment No. 168, dated July 8, 2003. In Amendment No. 168, the NRC staff concluded, in part, that NMC had appropriately identified all sources of uncertainty for reactor power level and "that the uncertainty determinations had been performed correctly. The NRC staff concluded that the uncertainty methodology the licensee used to calculate the KNPP power level instrument uncertainty was acceptable. The Westinghouse document used to determine the instrument uncertainty is WCAP-15591, Revision 1, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology - Kewaunee Nuclear Power Plant (Power Uprate to 1757 MWt-NSSS Power with Feedwater Venturis, or 1780 MWt-NSSS Power with Ultrasonic Flow Measurements, and 54F Replacement Steam Generators)." The supporting uncertainty analysis WCAP-15591 and explanatory calculations were performed at KNPP MUR power uprate conditions. The NRC staff used these documents and the July 8, 2003, SE as part of the basis of its review of the 6.0-percent stretch power uprate.

The NRC staff reviewed the licensee's determination of instrument uncertainties and setpoint determinations for those systems affected by the proposed 6.0-percent stretch power uprate.

Instrumentation and control systems are provided (1) to control plant processes having a significant impact on plant safety, (2) to initiate the reactivity control system (control rods), (3) to initiate the engineered safety features (ESF) systems and essential auxiliary supporting systems, and (4) to achieve and maintain a safe shutdown condition of the plant. Diverse instrumentation and control (I&C) systems and equipment are provided for the express purpose of protecting against potential common-mode failures of instrumentation and control protection systems. The NRC staff conducted a review of the reactor trip system, ESF actuation system (ESFAS), safe shutdown systems, control systems, and diverse instrumentation and control systems for the proposed 6.0-percent stretch power uprate to ensure that the systems and any changes required for the proposed stretch power uprate are adequately addressed such that the systems will continue to meet their safety functions following implementation of the proposed stretch power uprate. The NRC staff's review is also conducted to ensure that failures of the systems do not affect safety functions. The NRC's acceptance criteria related to the quality of design of protection and control systems are based on 10 CFR 50.55a(a)(1), 10 CFR 50.55a(h), and GDCs 1, 4, 13, 19, 20, 21, 22, 23, and 24. Specific review criteria are contained in SRP Sections 7.0, 7.2, 7.3, 7.4, 7.7, and 7.8.

Nuclear power plants are licensed to operate at a specified core thermal power level. The uncertainty values of this specified power level determines the probability of exceeding the power levels assumed in the design-basis transient and accident analysis. Furthermore, the determination of reactor safety trip setpoints is performed to ensure that sufficient allowance exists between the trip setpoint and the safety limit to account for instrument uncertainties. The NRC's regulatory requirements pertinent to this review can be found in 10 CFR 50.36(c)(1)(ii)(A), which requires that, where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting be so chosen that automatic protective action will correct the most severe abnormal situation anticipated without exceeding a safety limit. Limiting safety system settings are settings for automatic protective devices related to those variables having significant safety functions. Setpoints found to exceed TS limits are considered a malfunction of an automatic safety system. Such an occurrence could challenge the integrity of the reactor core, reactor coolant pressure boundary (RCPB), containment, and associated systems. Regulatory Guide (RG) 1.105, Revision 3, "Setpoint for Safety-Related Instrumentation," also provides relevant guidance for the NRC staff's review.

3.1.2 Technical Evaluation

3.1.2.1 Suitability of Existing Instruments

In its May 22, 2003, application, the licensee stated that no modifications to existing instrumentation and controls are required for the proposed 6.0-percent stretch power uprate, other than certain setpoints, plant process computer screen (PPCS) constants, and PPCS program changes. Based on this clarification, the NRC staff believes that the KNPP instrumentation and control systems will continue perform their intended functions, as required by the plant's license. The NRC staff's review concentrated on the instrument setpoint and setpoint methodology related to the proposed 6.0-percent stretch power uprate.

3.1.2.2 Instrument Setpoints

The proposed amendment reflects instrument setpoint changes consistent with the proposed 6.0-percent stretch power uprate. The NRC staff's evaluations of the setpoint changes for the identified instrumentation for the proposed power level are predicated on the assumption that analytical limits used by the licensee are based on the application of approved design codes.

3.1.2.2.1 RPS/ESFAS Instrumentation Trip Setpoint and Allowable Values

The Instrumentation, Systems, and Automation Society (ISA) Standard S67.04-1994, Part 1, "Setpoints for Nuclear Safety-Related Instrumentation," is endorsed by the NRC as an acceptable means for establishing setpoints and known contributing channel errors for the reactor protection system (RPS)/ESFAS instrumentation. The licensee cited this standard in its documentation to determine the channel uncertainty and setpoint determination. The NRC staff also used the following documents as part of its review of the proposed 6.0-percent stretch power uprate:

- WCAP-16040-P, "Power Uprate Project, Kewaunee Nuclear Power Plant, NSSS and BOP [Balance-of-Plant] Licensing Report," February 2003, section 6.8.

- WCAP-15821-P, Revision 0, "Westinghouse Protection Setpoint Methodology Kewaunee Nuclear Power Plant *(*Power Uprate to 1757 MWt-NSSS Power with Feedwater Venturis, or 1780 MWt-NSSS Power with Ultrasonic Flow Measurements, and 54F Replacement Steam Generators)," May 2003.

- KNPP General Nuclear Procedure, GNP-04.06.01, "Plant Setpoint Accuracy Calculation Procedure*,"* December, 2002.

3.1.2.2.2 KNPP Instrument Uncertainty Analysis and Determination

WCAP-16040-P discusses the uncertainty analysis used consistent with WCAP-11397, "Westinghouse Revised Thermal Design Procedure." WCAP-11397 was approved by the NRC staff in a January 17, 1989, safety evaluation report (SER). The methodology statistically combines the individual uncertainties using the "square root of the sum of the squares" method. The analysis includes the uncertainties for the method of measurement (i.e., RTDs, transmitters special test measurements), and the calibration of the instrumentation. Table 6.8-1 of WCAP-16040-P summarizes the initial condition uncertainties and shows that the calculated values are within those allowed by the safety analysis.

The NRC staff approved the KNPP uncertainty analysis and determination as part of its review of the KNPP MUR power uprate as discussed in Section 3.1.1 of this SE, which included a detailed review of WCAP-15591. Since the review included the uncertainties at a power level of 1780 MWt, which bounds the reactor power level of 1772 MWt requested in the May 22, 2003, application, the NRC staff concludes that licensee has appropriately identified the instrument uncertainties for a 6.0-percent stretch uprate from 1673 MWt to 1772 MWt.

3.1.2.2.3 KNPP Setpoint Methodology and Determination

The setpoint methodology used to determine plant setpoints has recently come under increased scrutiny as it has been determined that one of the methods (Method 3 in Part II of ISA Standard S67.04) may not provide adequate margin between the safety analysis limit (SAL), or analytical limit, and the allowable value, as required by 10 CFR 50.36. As such, the NRC staff issued a request for additional information (RAI) to the licensee in order to verify that adequate margin exists between these two values. In its response, the licensee asserted that TS values and Core Operating Limits Reports (COLR) values do not use allowable values, but instead use limiting safety system settings (LSSS) for TSs and setpoints for COLR values. The total instrument uncertainty is calculated with all measured and unmeasured instrument channel uncertainties. For increasing valued setpoints, the total instrument uncertainty, or channel statistical allowance, plus a margin, are subtracted from a defined SAL. The licensee then uses a more conservative plant setting to prevent a violation of the TS LSSS or COLR setpoint. Therefore, a margin of the total channel statistical allowance exists between the plant setting and the SAL. A tolerance band is established about the plant setting, that in the worse case, is within the band, thus ensuring that sufficient margin exists. The licensee's supplemental letter included a sample calculation of the overpower delta temperature trip function taken from WCAP-15821-P to illustrate the margin that exists. Finally, in its December 15, 2003, supplemental letter, the licensee stated that all setpoint calculations performed under the current program have been reviewed.

The NRC staff finds that the KNPP setpoint methodology used for the proposed 6.0-percent stretch power uprate is acceptable because of the demonstration that sufficient margin exists between the plant setting and the analytical limit, and thus meets the requirements of 10 CFR 50.36 and RG 1.105.

3.1.3 Conclusion

Based on the above evaluation, the NRC staff finds that the KNPP I&C systems will continue perform their intended functions and will continue to comply with the NRC's acceptance criteria related to the quality of design of protection and control systems that are based on, as defined in 10 CFR 50.55a(a)(1), 10 CFR 50.55a(h), and GDCs 1, 4, 13, 19, 20, 21, 22, 23, and 24. The NRC staff concludes that the licensee's instrument setpoint methodology for the proposed 6.0-percent stretch power uprate is consistent with the KNPP licensing basis and, therefore, is acceptable.

3.2 <u>Reactor Systems</u>

3.2.1 Regulatory Evaluation

The NRC staff's review in the area of reactor systems covers the impact of the proposed stretch power uprate on (1) fuel design, (2) nuclear design, (3) thermal-hydraulic design, (4) performance of control and safety systems connected to the reactor system and reactor coolant system (RCS), and (5) loss-of-coolant accident (LOCA) and non-LOCA transient analyses. The review is conducted to verify that the licensee's analyses bound plant operation at the LOCA stretch power level and that the results of the licensee's analyses related to the areas under review, will continue to meet the applicable acceptance criteria following implementation of the proposed stretch power uprate. The NRC staff used guidance and acceptance criteria for the reactor systems contained in Chapters 4, 5, 6, and 15 of

NUREG-0800. In KNPP License Amendment No. 167 the NRC staff approved the transition from Framatome ANP fuel to Westinghouse 422 VANTAGE + nuclear fuel with PERFORMANCE + features (422V+ fuel). Although the licensee had not requested a power uprate as part of the fuel transition amendment, the majority of the licensee's analyses and the NRC staff's review of the fuel transition amendment considered the proposed stretch power uprate level of 1772 MWt.

3.2.2 Technical Evaluation

3.2.2.1 Nuclear Steam Supply System Parameters

The nuclear steam supply system (NSSS) design parameters provide the RCS and secondary system conditions for use in NSSS analyses and evaluations. The licensee provided a listing of key plant parameters corresponding to the proposed stretch power uprate level of 1772 MWt in Table 2.1-1 in WCAP-16040. The major parameters include core power level, NSSS thermal power level, thermal design flow, reactor coolant pressure and temperatures, steam generator (SG) pressure, steam temperature, and steam flow rate. The major changes of these design parameters from the current values include increased core power level, increased minimum value for reactor coolant average temperature (T_{avg}), lower maximum steam pressure, lower maximum steam temperature, and a higher steam flow rate. The NRC staff evaluated these changes and found them to adequately represent the plant operating conditions at the proposed core power level of 1772 MWt. Also, these parameters are used in the licensee's safety analyses performed to support its proposed stretch power uprate, which resulted in acceptable margin to safety analysis limits. Therefore, the NRC staff finds the NSSS design parameters acceptable.

3.2.2.2 Reactor Coolant System

The changes in NSSS design parameters that impact the RCS design bases functions include the increase in core power and the allowable range for average RCS temperature (T_{avg}). The thermal design flow of 89,000 gpm/loop was maintained. The RCS temperature associated with the proposed stretch power uprate remains within the bounds of the original design temperature for the system. Sufficient core cooling under stretch power uprate conditions is verified by various plant transient and safety analyses. The NRC staff finds that the changes of RCS operating parameters associated with the stretch power uprate are acceptable based on the results of the safety analyses addressed in Section 3.2.2.12 of this SE.

3.2.2.3 Safety Injection System

The adequacy of the safety injection system (SIS) during the injection and sump recirculation phases following a LOCA was verified in the LOCA analyses performed at a core power level of 1772 MWt with acceptable results. For the non-LOCA events, the performance of the SIS was also verified by various safety analyses performed in support of the proposed stretch power uprate.

The licensee concluded that no system modifications are required to support the proposed stretch power uprate. The NRC staff agrees with the licensee's assessment based on the acceptable results of the safety analyses addressed in Section 3.2.2.12 of this SE.

3.2.2.4 Residual Heat Removal System

Operation at a higher power level increases the amount of decay heat being generated in the core, which results in a higher heat load to the residual heat removal (RHR) system for plant cooldown. The licensee evaluated the RHR system for the normal cooldown requirements at a core power level of 1772 MWt and found that the increased power level causes an increase to the cooldown time using the RHR system.

Normal cooldown is accomplished with two RHR heat exchangers and two component cooling water (CCW) heat exchangers in service. Several cooldown scenarios were analyzed to assess the impact of the stretch power uprate on cooldown time. Assuming design-bases RHR and CCW heat exchanger flows, and the original design service water (SW) system temperature of 66 ^{o}F, a core power uprate to 1772 MWt would increase the time for the RHR system to cool the plant down, from 350 ^{o}F to 140 ^{o}F to 17.5 hours. A total cooldown time of 21.5 hours with 4 hours of steam dump cooling would be required. However, at the current reduced RHR and CCW heat exchanger flows and the current maximum allowable SW system temperature of 80 ^{o}F of, a core power uprate to 1772 MWt would increase the RHR system cooldown time to 76.2 hours, and a total cooldown time of 80.2 hours with a required 4 hours of steam-dump cooling. The NRC staff finds that the increase of time for normal plant cooldown using RHR system is acceptable since there is no regulatory requirement for a maximum cooldown time during a normal plant shutdown.

3.2.2.5 Nuclear Steam Supply System Transients

In its stretch power uprate application, the licensee has evaluated the NSSS design transients to account for any impacts of the stretch power uprate. The NSSS design transients are traditionally developed for fatigue analyses of the various NSSS components using conservative assumptions. The licensee provided a tabulation comparing the plant operating conditions at the current power rating and the proposed NSSS power level of 1780 MWt. The licensee has evaluated the changes in the plant operating conditions and concludes that since the limiting values of primary and secondary system parameters are not changed, the existing design transients remain valid for the proposed stretch power uprate with one exception, the feedwater temperature is approximately 10 of higher than the original design and necessitates revision in the various design transients related to this parameter. The licensee has revised its NSSS design transients to incorporate this change. The NRC staff has reviewed the licensee's submittals with respect to NSSS design transients and finds it acceptable.

3.2.2.6 Fuel System Design Evaluation

NUREG-0800, Section 4.2 provides the basis for the NRC staff's requirements regarding the fuel system design. The objectives of the fuel system review are to provide assurance that: (1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOO), (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained.

The NRC staff's review covers fuel system damage mechanisms, limiting values for important parameters, and performance of the fuel system during normal operation, AOOs, and postulated accidents. The NRC's acceptance criteria are based on: (1) 10 CFR 50.46 for core cooling; (2) GDC-10 for assuring that specified acceptable fuel design limits (SAFDLs) are not

exceeded during any condition of normal operation, including AOOs; (3) GDC-27 for the reactivity control system being designed with appropriate margin, and in conjunction with the emergency core cooling system (ECCS), being capable of controlling reactivity and cooling the core under post accident conditions; and (4) GDC-35 for providing an ECCS to transfer heat from the reactor core following any loss of reactor coolant. Specific review criteria are contained in SRP Section 4.2.

The NRC staff previously reviewed and approved the licensee's analyses related to the fuel system design at the proposed stretch power uprate level of 1772 MWt as part of License Amendment No. 167, dated April 4, 2003, fuel transition amendment. The NRC staff reviewed the Westinghouse 422V+ fuel design and its compatibility with the current Framatome ANP fuel for the fuel transition period (mixed cores) through cores with all Westinghouse 422V+ fuel. The 422V+ fuel assembly has been designed to be compatible with the current Framatome ANP fuel, reactor internals and interfaces, fuel handling equipment, and refueling equipment. The licensee demonstrated that the 422V+ fuel design is both mechanically and hydraulically compatible with the Framatome ANP fuel, that adequate grid load margin exists such that core coolable geometry and control rod insertion requirements are satisfied, and that all 422 V+ fuel rod design criteria are satisfied at the stretch power uprate level of 1772 MWt. The NRC staff's review and approval of these elements is discussed in detail in Section 2.1 of the safety evaluation for License Amendment No. 167 fuel transition.

As part of the current review, the NRC staff requested clarification regarding the licensee's evaluation of the Framatome ANP fuel rod performance and design criteria at the proposed stretch power uprate level of 1772 MWt. The licensee provided additional information in its December 15, 2003, submittals letter to demonstrate that the design basis for the Framatome ANP fuel is satisfied for KNPP at the stretch power uprate conditions. The licensee provided a description of each applicable criterion, the evaluations performed and the results of the evaluations. The licensee's evaluations incorporated approved methods, and the results demonstrate that all fuel performance and design acceptance criteria are satisfied. Based on this, the NRC staff finds that all Framatome ANP fuel rod design criteria are satisfied at the stretch power uprate level of 1772 MWt.

The NRC staff has reviewed the licensee's analyses related to the effects of the stretch power uprate on the fuel system design. The NRC staff concludes that the licensee has adequately accounted for the effects of the stretch power uprate on the fuel system and demonstrated that: (1) the fuel system will not be damaged as a result of normal operation and AOOs; (2) the fuel system damage will never be so severe as to prevent control rod insertion when it is required; (3) the number of fuel rod failures will not be underestimated for postulated accidents; and (4) coolability will always be maintained. Based on this, the NRC staff concludes that the fuel system and associated analyses will continue to meet the requirements of 10 CFR 50.46, GDC-10, GDC-27, and GDC-35 following implementation of the stretch power uprate. Therefore, the NRC staff finds the proposed stretch power uprate acceptable with respect to the fuel system design.

3.2.2.7 Nuclear Design Evaluation

The NRC staff reviews the nuclear design of the fuel assemblies, control systems, and reactor core to ensure that fuel design limits will not be exceeded during normal operation or anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. The

NRC staff's review covers core power distribution, reactivity coefficients, reactivity control requirements and control provisions, control rod patterns and reactivity worth, criticality, burn-up, and vessel irradiation. The NRC's acceptance criteria are based on: (1) GDC-10 for assuring that SAFDLs are not exceeded during any condition of normal operation, including AOOs; (2) GDC-11 for the core design to assure that the prompt inherent nuclear feedback characteristics compensate for a rapid increase in reactivity; (3) GDC-12 for precluding or detecting and suppressing power oscillations which could result in conditions exceeding SAFDLs; (4) GDC-13 for instrumentation and controls to monitor variables and systems affecting the fission process over anticipated ranges for normal operation, AOOs and accident conditions, and maintaining the variables and systems within prescribed operating ranges; (5) GDC-20 for automatic initiation of the reactivity control systems to assure that SAFDLs are not exceeded as a result of AOOs and to assure automatic operation of systems and components important to safety under accident conditions; (6) GDC-25 for a single malfunction of the reactivity control system to not cause a violation of the SAFDLs; (7) GDC-26 for providing two independent reactivity control systems of different designs, and each system having the capability to control the rate of reactivity changes resulting from planned, normal power changes; (8) GDC-27 for the capability of the reactivity control systems in conjunction with poison addition by the ECCS to reliably control reactivity changes under postulated accident conditions, with appropriate margin for stuck rods; and (9) GDC-28 for the effects of postulated reactivity accidents neither resulting in damage to the RCPB greater than limited local yielding, nor causing sufficient damage to impair significantly the capability to cool the core. Specific review criteria are contained in SRP Section 4.3.

The NRC staff previously reviewed and approved the licensee's analyses related to the nuclear design evaluations at the proposed stretch power uprate level of 1772 MWt as part of the fuel transition amendment dated April 4, 2003. The licensee utilized an NRC-approved methodology to evaluate the nuclear design of the Framatome ANP and Westinghouse 422V+ fuel designs for transition cores through an all 422V+ core at the proposed stretch power uprate level of 1772 MWt. The NRC staff's review and approval of these elements is discussed in detail in Section 2.2 of the SE for License Amendment No. 167.fuel transition SER.

The NRC staff has reviewed the licensee's analyses related to the effect of the proposed stretch power uprate on the nuclear design. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed stretch power uprate on the nuclear design and has demonstrated that the fuel design limits will not be exceeded during normal or anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. Based on this evaluation and in coordination with the reviews of the fuel system design, thermal and hydraulic design, and transient and accident analyses, the NRC staff concludes that the nuclear design of the fuel assemblies, control systems, and reactor core will continue to meet the applicable requirements of GDCs 10, 11, 12, 13, 20, 25, 26, 27, and 28 following implementation often proposed stretch power uprate. Therefore, the NRC staff finds the proposed stretch power uprate acceptable with respect to the nuclear design.

3.2.2.8 Thermal and Hydraulic Design Evaluation

The NRC staff reviews the thermal and hydraulic design of the core and the RCS to confirm that the design: (1) has been accomplished using acceptable analytical methods; (2) is equivalent to or a justified extrapolation from proven designs; (3) provides acceptable margins of safety from conditions which would lead to fuel damage during normal reactor operation and

AOOs; and (4) is not susceptible to thermal-hydraulic instability. The NRC's acceptance criteria are based on GDC-10 for the reactor core being designed with appropriate margin to assure that SAFDLs are not exceeded during normal operation or AOOs. Specific review criteria are contained in SRP Section 4.4.

The NRC staff previously reviewed and approved the licensee's analyses related to KNPP core thermal and hydraulic design for power levels up through an MUR uprate power level of 1673 MWt. The NRC staff reviewed and approved the licensee's evaluations and application of departure from nucleate boiling (DNB) methodologies, hydraulic compatibility of the current Framatome and Westinghouse fuel types, and thermal-hydraulic related transition core effects. The NRC staff's review and approval of these elements for KNPP power levels up through 1673 MWt is discussed in detail in Section 2.3 of the SE for License Amendment No. 167. In that SE, for the stretch power uprate level of 1772 MWt, the NRC staff stated that the licensee will reevaluate the DNB ratio (DNBR) penalties and design limit DNBR values, and appropriate changes will be submitted to the NRC as part of the licensee's application for the stretch power uprate. The licensee's submittals dated May 22, 2003, application discussed three proposed changes to accommodate the proposed stretch power uprate. The three proposed changes include (1) a reduction in the design limit DNBR value from 1.24 to 1.23, (2) a reduction in rod bow penalty from 2.6 percent to 0 percent for operating Cycle 26 only, and (3) a Cycle 26 transition core DNBR penalty of 2.5 percent. In a January 30, 2004, letter, the licensee subsequently withdrew the proposed reduction in rod bow penalty and stated that KNPP would be applying the standard Westinghouse rod bow methodology, including application of a 2.6 percent rod bow DNBR penalty to KNPP Cycle 26 core thermal-hydraulic analyses at the stretch uprate conditions.

The NRC staff reviewed and approved the licensee's application of the Westinghouse Revised Thermal Design Procedure (RTDP) and associated instrumentation uncertainty values calculated in WCAP-15591 as part of License Amendment No. 167. For the proposed stretch power uprate, the licensee proposes to revise the design limit DNBR value previously calculated using this approved methodology, from 1.24 to 1.23. The licensee stated in the letter dated November 5, 2003, supplemental letter, that the basis for this change is the use of the actual calculated instrument uncertainties in WCAP-15591, rather than previously applied conservative instrumentation uncertainty assumptions. The NRC staff finds this change to be acceptable because the licensee applies the previously approved methodology RTDP and instrument uncertainty values (WCAP-15591) to calculate the revised design limit DNBR value of 1.23. Additionally, the licensee is not revising the safety analysis DNBR limit of 1.34, which accounts for rod bow, instrument uncertainty and transition core DNBR penalties.

To support the proposed stretch power uprate, the licensee reevaluated the transition core DNBR penalty to be applied during the fuel transition cycles. The licensee extended the analyses performed to support the KNPP fuel transition by evaluating several additional transition core configurations. The licensee provided the results of these analyses in a figure showing transition core DNBR penalty as a function of Westinghouse 422V+ fuel loaded in the core in its November 5, 2003, supplemental letter. These additional analyses satisfied NRC staff concerns raised in the fuel transition amendment review regarding variance in curve fitting. The licensee applied an NRC-approved methodology (WCAP-11837-P-A, "Extension of Methodology for Calculating Transition Core DNBR Penalties," dated January 1990) to calculate the transition core DNBR penalties, and for Cycle 26, the licensee will apply a conservative transition core DNBR penalty of 2.5 percent. The transition core DNBR penalty will be reduced as more Westinghouse 422V+ fuel assemblies are loaded in the KNPP core in future operating

cycles, and eventually is eliminated for an all Westinghouse 422V+ core. The NRC staff finds that the transition core DNBR penalty is a conservative and reasonable value, and is acceptable because the licensee applied approved methods to determine the KNPP-specific penalties.

The NRC staff has reviewed the licensees analyses related to the effects of the proposed stretch power uprate on the thermal and hydraulic design of the core and the RCS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed stretch power uprate on the thermal and hydraulic design and demonstrated that the design (1) has been accomplished using acceptable analytical methods, is (2) equivalent to proven designs, (3) provides acceptable margins of safety from conditions which would lead to fuel damage during normal reactor operation and AOOs, and (4) is not susceptible to thermal-hydraulic instability. Based on this, the NRC staff concludes that the thermal and hydraulic design will continue to meet the requirements of GDC-10 following implementation of the proposed stretch power uprate. Therefore, the NRC staff finds the proposed stretch power uprate acceptable with respect to thermal and hydraulic design.

3.2.2.9 Functional Design of Control Rod Drive System

The NRC staff's review covers the functional performance of the control rod drive system (CRDS) to confirm that the system can effect a safe shutdown, respond within acceptable limits during AOOs, and prevent or mitigate the consequences of postulated accidents. The review also covers the CRDS cooling system to ensure that it continues to meet its design requirements. The NRC's acceptance criteria are based on: (1) GDC-4 for the environmental conditions caused by a high energy line break (HELB) or moderate energy line break during normal plant operation, as well as postulated accidents; (2) GDC-23 for failing into a safe state; (3) GDC-25 for the functional design of redundant reactivity control systems to assure that SAFDLs are not exceeded for malfunction of any reactivity control systems; (4) GDC-26 for the capability of the reactivity control systems to regulate the rate of reactivity changes resulting from normal operations and AOOs; (5) GDC-27 for the combined capability of reactivity control systems and the ECCS to reliably control reactivity changes to assure the capability to cool the core under accident conditions; (6) GDC-28 for postulated reactivity accidents; and (7) GDC-29 for functioning under AOOs. Specific review criteria are contained in SRP Section 4.6.

The NRC staff verifies the safe shutdown and consequences of postulated accidents as part of the Nuclear Design Evaluation and review of the plant transient and accident analyses.

The NRC staff previously reviewed and approved the licensees analyses related to the rod cluster control assembly (RCCA) insertion at the stretch power uprate level of 1772 MWt as part of License Amendment No. 167. The design criterion of concern is RCCA drop time. The licensee performed a drop time analysis under worst-case conditions, and considering the 422V+ fuel transition and the proposed stretch power uprate of 1772 MWt core power. The licensee calculated the maximum RCCA drop time with seismic allowance to be 1.59 seconds, which satisfies the KNPP TS limit of 1.80 seconds. Therefore, the NRC staff found that the RCCA drop times are acceptable.

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed stretch power uprate on the functional design of the CRDS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed stretch power uprate on the system and demonstrated that the system's ability to effect a safe shutdown, respond within acceptable limits, and prevent or mitigate the consequences of postulated accidents will be

maintained following the implementation of the proposed stretch power uprate. Based on this, the NRC staff concludes that the fuel system and associated analyses will continue to meet the requirements of GDCs 4, 23, 25, 26, 27, 28, and 29 following implementation of the proposed stretch power uprate. Therefore, the NRC staff finds the proposed stretch power uprate acceptable with respect to the functional design of the CRDS.

3.2.2.10 Overpressure Protection During Power Operation

Overpressure protection for the RCPB during power operation is provided by safety relief valves and the RPS. The NRC's acceptance criteria are based on: (1) GDC-15 for the RCS and associated auxiliary, control, and protection systems being designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs, and (2) GDC-31 for the RCPB being designed with sufficient margin to assure that it behaves in a nonbrittle manner and that the probability of rapidly propagating fracture is minimized. Specific review criteria are contained in SRP Section 5.2.2.

The NRC staff has reviewed the licensee's safety analyses related to the effects of the proposed stretch power uprate on the overpressure protection capability of the plant during power operation. Based on the acceptable results of the safety analyses for heatup events, the NRC staff concludes that the overpressure protection features will continue to provide adequate protection to meet GDC 15 and GDC 31 at an uprated core power level of 1772 MWt. Therefore, the NRC staff finds the proposed stretch power uprate acceptable with respect to overpressure protection during power operation.

3.2.2.11 Overpressure Protection During Low Temperature Operation

3.2.2.11.1 Vessel Fluence

Vessel fluence was calculated by Westinghouse using the DOORS 3.1 discrete ordinates code package (DOORS 3.1, "One- Two- and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System" Radiation Safety Information and Computation Center (RSICC) Computer Code Collection CCC-650, Oak Ridge National Laboratory, August 1996) with the BUGLE-96 cross section library (BUGLE-96, "Coupled 47 Neutron, 20 Gamma Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," RSICC Data Library Collection-185, Oak Ridge National Laboratory, March 1996). The calculations were based on the synthesis method in the (r, θ) and (r, z) planes and on an S_{16} order of angular quadrature and a P_5 expansion of the scattering cross section. The neutron sources were calculated from cycle-specific power distributions for the first 25 cycles. The proposed stretch power uprate initiates with Cycle 26, and it is assumed that subsequent cycles will retain the same power profile. The May 22, 2003, application provides the results of comparison of neutron flux and fluence calculations of surveillance capsules, stating that the results are within the limits prescribed in RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" US Nuclear Regulatory Commission, March 2001). Fluence values are reported for 33 and 54 effective full power years (EFPYs) of operation, corresponding to end of the current license and license extension, respectively. The methodology applied in the estimation of the vessel fluence follows the guidance in RG 1.190, and therefore, it is acceptable.

3.2.2.11.2 Pressure Temperature Limit Curves

The current pressure-temperature (PT) limit curves are discussed in WCAP-14278, Revision 1, "Kewaunee Heatup and Cooldown Limit Curves for Normal Operation," September 1998. Comparison of the fluence values (and corresponding adjusted reference temperature (ART) to the end of the current license (and projections to the end of an extended license) indicate that the proposed stretch power uprate will result in higher values. There are two options available to the licensee, (1) recalculating the PT curves for the new fluence values or (2) adjusting the range of validity and revising the PT curves upon expiration of the present PT limits. The licensee chose the latter one by reducing the expiration time in proportion to the ratio of the old fluence to the new fluence at the uprated power level. The new expiration time is 31.1 EFPYs from 33.0 EFPYs. At 31.1 EFPYs with the proposed stretch power uprate, the accumulated irradiation is equal to that at 33.0 EFPYs without the stretch power uprate. Because the relevant material properties depend upon the amount of irradiation, the method of arriving at the revised range of applicability is reasonable, and therefore, is acceptable.

3.2.2.11.3 Low Temperature Overpressure Protection (LTOP)

The stretch power uprate does not affect the 10 CFR Part 50, Appendix G related material properties. Therefore, the current LTOP limit setting will be valid as long as the fluence value remains below the value used in the calculation of the current limits, which is 31.1 EFPYs. The current limit setting will need to be revised at or before it reaches 31.1 EFPYs with the stretch power uprate (i.e., along with the PT limit curves). As in the case of the PT curves, the current LTOP setting is acceptable for 31.1 EFPYs.

3.2.2.11.4 Pressurized-Thermal Shock

The pressurized-thermal shock (PTS) requirements are described in 10 CFR 50.61 in terms of screening temperatures for RT_{PTS} of 270 °F for plates and axial welds and 300 °F for circumferential welds in the belt region. RT_{PTS} is defined in 10 CFR 50.61. The method described in 10 CFR 50.61 in conjunction with RG 1.99 and RG 1.190 was used for all of the beltline materials in KNPP except for the intermediate to lower shell circumferential weld. For this weld, the licensee applied and was granted an exemption to use the master curve methodology (WCAP-15075, "Master Curve Strategies for Reactor Pressure Vessel Assessment," September 1998, and from the requirements of 10 CFR Part 50, Appendices G and H, and 10 CFR 50.61, dated February 21, 2001).

Using the master curve methodology, the licensee reevaluated RT_{PTS} for the intermediate to lower shell circumferential weld and fluence values corresponding to 33 and 53 EFPYs, including the proposed stretch power uprate. The result demonstrated that this weld remains within the required limits of 10 CFR 50.61. The methodology and the results have been approved by the NRC in the February 21, 2001, Exemption.

3.2.2.12 Transient and Accident Analyses

The licensee reanalyzed the LOCA and non-LOCA transients and accidents in support of the proposed stretch power uprate. These analyses were performed at a rated core power of 1772 MWt, incorporated plant parameter values for those operating conditions, and considered the fuel transition by evaluating both Framatome ANP and Westinghouse 422V+ cores. The NRC staff previously reviewed and approved many of the licensee's transient and accident

analyses at the stretch power uprate level of 1772 MWt as part of License Amendment No. 167. The NRC staff's review of the LOCA and non-LOCA transients and accidents is discussed below.

3.2.2.12.1 LOCA Analyses

3.2.2.12.1.1 Large-Break LOCA and Small-Break LOCA Analyses

The licensee performed KNPP large-break loss-of-coolant accident (LBLOCA) and small-break loss-of-coolant accident (SBLOCA) reanalyses for cores containing Westinghouse Vantage+ (ZIRLO clad) fuel and Framatome 14x14 (Zircaloy clad) fuel assuming 102 percent (1807 MWt) of a core power of 1772 MWt. The licensee has also indicated that it will incorporate Westinghouse Vantage+ (ZIRLO-clad) fuel into upcoming Kewaunee cores. For at least the first operating cycle with the Vantage+ fuel, KNPP will be operated with a mixed core configuration with Framatome fuel. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," dated April 1995, describes Vantage+ fuel, and Appendices F and G to that document describe performing LOCA analyses for cores containing Vantage+ fuel using Westinghouse SBLOCA and LBLOCA methodologies. The licensee used the Westinghouse best-estimate LBLOCA analysis methodology described in WCAP-14449-P-A, Revision 1, "Westinghouse Best-Estimate LBLOCA Methodology", dated October 1999, to perform the LBLOCA analyses. This methodology is specifically applicable to Westinghouse pressurized-water reactors (PWRs) designed with upper plenum injection (UPI). The methodology also applies to this class of plants for LBLOCA analysis of Vantage+ fuel (WCAP -13677-P-A, "10 CFR 50.46 Evaluation Model Report", dated April 1993). The licensee used the Westinghouse "COSI" SBLOCA analysis (Appendix K) methodology described in WCAP-10054-P-A, Addendum 2, Revision 1, "Westinghouse SBLOCA Evaluation Methodology", dated July 1997. This methodology is applicable to Westinghouse PWR designs, including Westinghouse PWRs designed with UPI. These Westinghouse LOCA methodologies are relatively new, and have few limitations placed on their application. Westinghouse internal quality assurance processes provide guidance for analysts which address the usage limitations. The licensee identifies, evaluates, and reports updates to the analyses and methodologies for KNPP, as required by 10 CFR 50.46(a)(3).

The NRC staff previously reviewed and approved the licensee's analyses related to the LBLOCA and SBLOCA for KNPP at the stretch power uprate level of 1772 MWt as part of KNPP the fuel transition amendment. The licensee utilized NRC approved methodologies to evaluate these events and demonstrated that all acceptance criteria are satisfied. The NRC staff's review and approval of these transients for KNPP is discussed in detail in Section 2.4.1 of the SE for License Amendment No. 167.

The NRC staff has reviewed the licensee's analyses of the LBLOCA and SBLOCA and concludes that the licensee's analyses have adequately accounted for the changes in core design required for operation of the plant at the proposed stretch power uprate level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the SAFDLs are not exceeded. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of 10 CFR 50.46 following implementation of the proposed stretch power uprate. Therefore, the NRC staff finds the proposed stretch power uprate acceptable with respect to the LOCA analyses.

3.2.2.12.1.2 Post LOCA Long-Term Cooling

The regulatory requirement for long-term cooling (LTC) is provided in 10 CFR 50.46(b)(5), which states, "After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core." In practice, following successful calculated blowdown, refill, and reflood after initiation of a LOCA, the LTC requirement will be met if the fuel cladding remains in contact with water so that the fuel cladding temperature remains essentially at or below the saturation temperature. A potential challenge to LTC is that boric acid (H_3BO_3) could accumulate within the reactor vessel, precipitate, and block water needed to keep the fuel cladding wetted by water.

As reported in the License Amendment No. 168, the licensee previously performed LOCA and related analyses for a power level of 1772 MWt, with an added 0.6 percent uncertainty, for a total power of 1782 MWt. However, the licensee's request at that time was for a power of 1673 MWt. In the case of the long-term cooling analyses, the NRC staff limited its evaluation to LBLOCA and it found the long-term core cooling to be acceptable for the power uprate to 1673 MWt. The NRC staff did not evaluate the licensee's long-term cooling analyses for a power level of 1772 MWt, nor did it address other LOCA sizes since the previously approved analyses were deemed sufficiently close to the 1673 MWt power level that an evaluation was unnecessary.

The KNPP is a UPI design, where the low head emergency core cooling pumps deliver flow directly into the upper plenum. For this reason, the hot-leg switchover procedure that is applied to typical three-loop and four-loop Westinghouse designs to ensure long-term coolant injection above the core is not applied to KNPP. This approach has been found to be adequate for the historic long-term cooling requirement based upon post-accident response to a LBLOCA when operating at or near the originally analyzed power level. However, significant power increases have led the NRC staff to conclude that a more comprehensive evaluation is necessary to reasonably ensure compliance with 10 CFR 50.46(b)(5) because of the potential rate at which H_3BO_3 can accumulate in the reactor vessel. Consequently, the NRC staff reviewed the licensee's approach to control H_3BO_3 during LTC at the proposed stretch power uprate level of 1782 MWt.

The NRC staff addressed potential H_3BO_3 accumulation in the KNPP UPI design by considering three types of accidents: (1) LOCAs that rapidly depressurize the RCS to significantly less than the low pressure injection (LPI) pump discharge pressure, (2) LOCA's where RCS pressure remains above LPI discharge pressure and there is continued natural circulation (NC) of liquid through the RCS, and (3) LOCA's where RCS pressure remains above LPI discharge pressure and NC of liquid through the RCS is lost.

The licensee addressed Item 1 LOCAs in a letter from Thomas Coutu, "Kewaunee Nuclear Power Plant, Docket No. 50-305, License No. DPR-43, NMC Responses to NRC Request for Additional Information Concerning License Amendment Request No. 187 to the Kewaunee Nuclear Power Plant Technical Specifications (TAC No. MB5718)," Letter to NRC from Site Vice-President, NRC-03-016, February 27, 2003, when it addressed LBLOCAs as follows:

> For cold-leg breaks, the large volume of low-head safety injection (LHSI) flow to the upper plenum will make up for boil-off and metal heat and will establish reverse flow through the core.
>
> Similarly, for hot-leg breaks, the large volume of LHSI flow to the upper plenum will make up core boil-off, metal heat and entrainment out the break and natural circulation will preclude boron buildup.
>
> The Westinghouse long-term cooling methodology assumes that licensing basis ECCS flow is not adversely affected by the switchover from injection phase to sump recirculation.

The licensee concluded that LBLOCAs are not of concern.

LBLOCAs will often result in significant boiloff during reactor vessel refill/reflood that could concentrate H_3BO_3 in the reactor vessel. The NRC staff determined that the large LPI pump flow rate which immediately follows reflood will flush any concentrated H_3BO_3 solution out of the core due to the large-break flow rate for cold-leg breaks and due to reactor vessel mixing in combination with the large-break flow rate for hot-leg breaks. This will thereafter maintain H_3BO_3 at a sufficiently low concentration that there is no longer a concern.[1] Consequently, the NRC staff finds the licensee's LBLOCA conclusion to be acceptable.

Where RCS pressure remains above LPI discharge pressure and there is continued natural circulation (NC) of liquid through the reactor vessel and SG tubes, NC should provide sufficient mixing that H_3BO_3 concentration in the core will not be significantly higher than in the remainder of the RCS. Further, the combination of break and ECCS flow will prevent significant H_3BO_3 concentration in the RCS.[2] This type of LOCA is of no concern with respect to significant H_3BO_3 concentration.

Where RCS pressure remains above LPI discharge pressure and NC is lost, NC may be reestablished or it may be lost for a long time. In response to an NRC staff question, the licensee stated the following in its supplemental letter dated November 5, 2003:

> For the (small-break LOCA) scenario where the RCS can be refilled, natural circulation will be established and boron precipitation cannot occur.

This response does not address the concentration of H_3BO_3 that can occur prior to reestablishing effective NC nor does it address whether H_3BO_3 precipitation could occur when NC first moves water into the core. Therefore, the NRC staff has elected to combine consideration of this case with the case where NC is lost for a long time.

For the case where RCS pressure remains above LPI discharge pressure for an extended time and NC is lost, the licensee presented a rationale wherein it assumed 35 psia and showed that there were 18 hours in which it could establish effective long-term cooling. In response to an

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[1] There is more concern in plants where LPI is into the cold-legs because LPI flow does not pass through the core to reach the break. The NRR staff intends to assess this behavior further as part of a generic investigation.

[2] In many of these LOCAs, there will be no boiling in the core, and therefore, no mechanism for increasing H_3BO_3 concentration beyond that of the RCS and the refueling water storage tank.

NRC staff question, the licensee stated the following in its supplemental letter dated November 5, 2003:

> The 35 psia assumption referred to is not really a containment pressure assumption. Rather it is a recognition that at 35 psia RCS backpressure, the low head pumps will be injecting into the upper plenum.... The Emergency
>
> Operating Procedures (EOP) direct the operator to establish low-head recirculation (via the upper head injection lines) at RCS pressures below 150 psig, and confirm a minimum flow (1500 gpm). Therefore, basing the boric acid solubility limit on a pressure well below 150 psig is justified.

Although the NRC staff agrees that there is conservatism associated with 150 psig in comparison to 35 psia, this is misleading. If the assumed backpressure is higher than that encountered during cooldown, and the H_3BO_3 concentration achieved is higher than the H_3BO_3 saturation concentration that will encountered during cooldown, then there is a potential for H_3BO_3 precipitation. Therefore, in an acceptable analysis model, the NRC staff will require that time to initiate effective hot side injection be based upon the minimum H_3BO_3 saturation concentration condition that will be encountered during cooldown unless a higher H_3BO_3 concentration is justified. These, and other modeling assumptions that have not been justified, are not unique to this licensee, and the NRC staff has previously questioned H_3BO_3 behavior modeling during long term cooling when reviewing applications from other licensees. In these cases, the NRC staff has elected to address the concerns on a generic basis and, as is illustrated in the September 29, 2003, letter from Pham, B. M., "Palo Verde Nuclear Generating Station, Unit 2 (PVNGS-2) - Issuance of Amendment on Replacement of Steam Generators and Uprated Power Operations (TAC No. MB3696)," to Gregg R. Overbeck, Arizona Public Service Company, ML032720538, September 29, 2003, the NRC staff has completed an interim evaluation by comparing long-term cooling characteristics with cases where effective H_3BO_3 dilution action was initiated well before the NRC staff judged the action was necessary.

These characteristics, taken from the September 29, 2003, Palo Verde letter, are compared to the Kewaunee characteristics provided in the licensee's supplemental letter dated February 9, 2004 in Table A below.

Table A - Comparison of H_3BO_3 Accumulation Characteristics

	Characteristic	Byron/ Braidwood 5% uprate	ANO-2 7.5% uprate	Palo Verde 2.94% uprate	Kewaunee 6% uprate (7.4% including previous uprate)
1	Time to reach H_3BO_3 saturation (hours)	8.53 (5/4/01) 6.0 (4/12/02)	~2.4 to 7.3, depending on assumptions	~3.5 (FSAR)	7.8
2	Power (MWt)	3587	3026	4070	1772 + 0.6% uncertainty

Table A - Comparison of H_3BO_3 Accumulation Characteristics					
	Characteristic	Byron/ Braidwood 5% uprate	ANO-2 7.5% uprate	Palo Verde 2.94% uprate	Kewaunee 6% uprate (7.4% including previous uprate)
3	Decay heat generation rate multiplier (dimensionless).	1 (5/4/01) 1.2 (4/12/02)	1.1	1.1	1
4	Assumed H_3BO_3 saturation limit (wt%).	23.53	27.6	30	23.53
5	Core plus upper plenum volume below hot leg (ft^3).	1072*	940	Multiplying power by mixing volume ratio gives approximately ANO power	Power to volume ratio is similar between 2 and 4 loop Westinghouse plants
6	Time to hot-leg injection via emergency operating procedures (hours)	Consistent with Item 1 prediction	2 to 4	2 to 3	6.6
*Value is from NUREG-1269, "Loss of Residual Heat Removal System, Diablo Canyon Nuclear Power Plant, Unit 2, April 10, 1987," June 1987.					

The NRC staff notes that Byron/Braidwood procedures would reasonably ensure establishing effective hot-leg injection in 6 hours whereas the licensee would establish it in 6.6 hours; essentially the same time. Although the Combustion Engineering-designed plants would establish effective hot-leg injection in less time, the NRC staff judges there is significant unidentified conservatism that would lead to a time greater than 7.8 hours before H_3BO_3 precipitation would jeopardize core cooling. Further, there is a low probability that conditions leading to significant H_3BO_3 accumulation will be encountered. Therefore, while the NRC staff cannot endorse the licensee's evaluation, the NRC staff believes, on an interim basis, that there is sufficient basis to approve the license amendment with respect to long-term cooling and potential H_3BO_3 precipitation concerns.

This NRC staff's conditional acceptance will remain effective until generic concerns associated with long-term cooling are rectified, at which time, the licensee will have to establish that it is in compliance with the resolution of the generic concerns.

3.2.2.12.1.3 RCS Flow Rate

In the License Amendment No. 168, the NRC staff concluded that the licensee's determination of RCS flow rate was acceptable for purposes of the 1.4-percent MUR power uprate. However, the NRC staff also identified a contribution to the determination of RCS flow rate that did not appear to be addressed by the licensee, and the NRC staff did not assess the effect of letdown,

makeup, reactor coolant pressure (RCP) cooling, RCP seal injection, and the pressurizer on the prediction of RCS flow rate. The NRC staff concluded that a more complete review of these areas was necessary for review of the licensee's 6.0-percent power uprate.

In License Amendment No. 168, the NRC staff concluded that the licensee used the equivalent of the following equation for determination of RCS flow rate[3]:

$$W = \{Q_{SG} - Q_P + Q_L\} / (h_H - h_C) \tag{1}$$

where:

where:	W	=	flow rate
	Q_{SG}	=	calorimetrically-determined steam generator thermal output
	Q_P	=	reactor coolant pump (RCP) heat addition rate
	Q_L	=	RCS net heat loss rate
	h_H	=	hot-leg enthalpy (determined at T_H and P)
	h_C	=	cold-leg enthalpy (determined at T_C and P)
	T_C	=	cold-leg temperature (nominal value = 542.1°F)
	T_H	=	hot-leg temperature (nominal value = 608.5°F)
	P	=	pressurizer pressure (nominal value = 2250 psia)

The nominal values correspond to a 1757 MWt NSSS power. Note that Q_{SG} and Q_P are associated with heat outside of the reactor vessel whereas Q_L includes all heat loss. Yet $h_H - h_C$ is the enthalpy change between the locations of measurement of T_C and T_H, which includes sections of the hot and cold-leg pipes and the reactor vessel. This raised a question of consistency in treating the terms.

The NRC staff determined that a heat balance over the reactor vessel and the portion of the hot and cold-leg pipes located between T_h and T_c yielded the following equation with the assumption that the only mass flow was at the locations of T_h and T_c:

$$W = (Q_{core} - Q_{loss\Delta T}) / (h_h - h_c) \tag{2}$$

where $Q_{loss\Delta T}$ is the heat loss rate between the locations of T_h and T_c.

A straightforward RCS heat balance shows that:

$$Q_{SG} = Q_{core} + Q_P - Q_L \tag{3}$$

Equations 2 and 3 combine to yield:

$$W = \{Q_{SG} - Q_P + Q_L - Q_{loss\Delta T}\} / (h_H - h_C) \tag{4}$$

The $Q_{loss\Delta T}$ term does not appear in the Equation 1. The NRC staff therefore determined that, if its analysis was correct, the licensee over-predicted the flow rate when it used Equation 1. The NRC staff estimated the over-prediction to be about 150 gpm.

The NRC staff further concluded that:

- The licensee over-predicted W by 26 gpm due to determining h_H and h_C at pressurizer pressure as opposed to determination at the T_H and T_C locations.

[3] Dimensional information and the number of loops have been removed for simplicity.

- Letdown and makeup are included as total values without consideration of which loops are actually affected.

- Pressurizer spray flow and pressurizer surge line flow are included without consideration of which loops are actually affected.

- The NRC staff's analysis was developed with an assumption that there was no mass flow into or out of the control volume, except for flow in the hot and cold legs, an assumption that may be inconsistent with the physical configuration associated with such items as letdown, makeup, RCP cooling, RCP seal injection, and the pressurizer.

The NRC staff believes that normal charging and the pressurizer spray line cross the boundary of the control volume between the T_H and T_C locations, but letdown, seal injection flow, the RCP thermal barrier cooler heat removal, and the pressurizer surge line are outside the heat balance boundary defined by the locations of T_h and T_c. For purposes of estimating $Q_{loss\Delta T}$, the NRC staff assumed the following characteristics:

- Normal charging minus seal injection = 50 gpm $\equiv$0.8 MBTU/hr
- Pressurizer spray = 15 gpm $\equiv$0.41 BTU/hr
- RV heat loss rate = 0.2 MBTU/hr
- Control rod drive heat loss rate = 1.5 MBTU/hr
- Hot and cold-leg pipe heat loss rate = 0.05 MBTU/hr

Normal charging introduces cool water from the regenerative heat exchanger downstream of T_c and, in effect, is a heat loss. Conversely, the pressurizer spray line does not influence T_c. Thus, $Q_{loss\Delta T}$ is approximately 0.8 + 0.2 + 1.5 + 0.05 = 2.6 MBTU/hr $\equiv$80 gpm.

RCS flow passes through the elbow tap measurement location before the letdown stream is removed. Assuming letdown and charging rates are identical, the same flow rate is reinjected into the RCPs and cold-legs, so that total RV flow is unaffected. Conversely, the nominal 15 gpm pressurizer spray bypasses the reactor vessel but is indicated by the elbow tap instrumentation.

The NRC staff consequently estimates that the licensee's analysis over-predicts RCS flow rate by 26 + 80 + 15 $\equiv$ 100 gpm. The licensee has committed to reduce its predicted flow rate by this amount when making a calorimetric calibration of its elbow tap flow meters (Reference 31). Since the MUR power uprate dated July 8, 2003, review was conducted with respect to a 1.4 percent power increase, a number of other considerations were identified that were not included in the NRC staff's review. They were addressed in the letter from Shukla, Girija S., "Diablo Canyon Power Plant, Unit No. 1 and Unit No. 2 - Issuance of Amendment - Revision of Technical Specification (TS) Table 3.3.1-1, 'Reactor Trip System Instrumentation,' and Revised Reactor Coolant System Flow Measurement) TAC Nos. MB6760 and MB6761," Letter from NRC to Gregory M. Rueger, Pacific Gas and Electric Company, ML032380158, August 21, 2003, review that was being conducted at the same time. These considerations include the following:

- Elbow tap flow meter coefficients remain sufficiently constant that the relative changes of flow rate through the cold-leg elbows can be correlated with the relative changes in the elbow tap ΔPs during long-term operation provided sufficient attention is paid to

calibration of instrumentation that translates ΔP to flow rate.

- RCS flow rate is likely to decrease by 0.6 to 1.0 percent during the first one or two cycles of operation, an effect that will be indicated by elbow tap ΔPs.

- Once calibrated, the elbow taps will reflect any RCS flow rate changes due to core fouling.

- A rough approximation, assuming elbow tap calibration is accomplished at the beginning of a fuel cycle, led the staff to conclude that the change in H_3BO_3 concentration over an operation cycle will cause indicated flow rate to decrease by about 100 gpm per loop. This introduces an error in the conservative direction during an operation cycle when rate calibrations are performed with the highest H_3BO_3 concentration.
- There is about a 0.8 percent to 1.2 percent change in RCS flow rate with a change in power from zero to 100 percent. This has a negligible effect on elbow tap calibration when power is greater than 80 percent.

These conclusions are applicable to the KNPP.

3.2.2.12.2 Non-LOCA Transients and Accidents

The licensee reanalyzed the Non-LOCA transients and accidents in support of the proposed stretch power uprate. These analyses were performed at a rated core power of 1772 MWt, incorporated plant parameter values for those operating conditions, and considered the fuel transition by evaluating both Framatome ANP and Westinghouse 422V+ cores. The NRC staff previously reviewed and approved many of the licensees transient and accident analyses for Westinghouse 422V+ core designs at the stretch power uprate level of 1772 MWt as part of License Amendment No. 167.

In the SE for License Amendment No. 167, the NRC staff stated that the licensee's submittals included transient and accident analyses and results for the proposed Westinghouse 422V+ fuel type only. For the fuel transition needed (not considering any power uprates), the Westinghouse fuel will be more limiting than the Framatome ANP fuel. The margin to DNBR for the Framatome ANP fuel will increase due to the increase in local flow caused by the mixed core effects and a decrease in $F_{\Delta H}$ due to the once-burned status. For the previously approved 1.4-percent MUR power uprate, the analytic margin to the DNBR limit for the Framatome ANP fuel should also increase for the same reasons as for the fuel transition only. Additionally, the MUR uprate relies on exchanging the power uprate for a decrease in power level uncertainty. However, for the stretch power uprate level of 1772 MWt, the power increase effects may not be overcome by the gained transition core DNBR margin. To support the stretch power uprate, the NRC staff requested that the licensee submit the Framatome ANP non-LOCA transient and accident analyses results. The licensee provided this information in its supplemental letter dated December 15, 2003, and a discussion is included in each of the transient and accident discussions below.

3.2.2.12.2.1 Excessive Heat Removal Due to Feedwater Temperature Reduction or Flow Increase

Excessive heat removal causes a decrease in moderator temperature, which increases core reactivity and can lead to a power level increase and a decrease in shutdown margin. Any unplanned power level increase may result in fuel damage or excessive reactor system pressure. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers postulated initial core and reactor conditions, methods of thermal and hydraulic analysis, sequence of events, assumed reactions of reactor system components, functional and operational characteristics of the RPS, required operator actions, and the results of the transient analyses. Acceptance criteria are based on GDC-10 as it relates to the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits SAFDLs are not exceeded during normal operations including AOOs; GDC-15 as it relates to the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operations, including AOOs; GDC-20 as it relates the RPS being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded during any condition of normal operation, including AOOs; and GDC-26 as it relates to the reliable control of reactivity changes to ensure that SAFDLs are not exceeded, including AOOs. Specific review criteria are contained in SRP Section 15.1.1-4.

The NRC staff previously reviewed and approved the licensee's analyses related to the excessive heat removal due to feedwater temperature reduction or flow increase for a Westinghouse 422V+ core at the stretch power uprate level of 1772 MWt as part of License Amendment No. 167 transition amendment. The licensee utilized NRC-approved methodologies to evaluate this transient and demonstrated that all acceptance criteria are satisfied. The NRC staff's review and approval of this transient for the Westinghouse 422V+ core is discussed in detail in Section 2.4.2 of the SE for License Amendment No. 167.

To support the stretch power uprate license amendment request, the NRC staff requested that the licensee submit the Framatome ANP fuel non-LOCA transient and accident analysis results. The licensee provided this information in its supplemental letter dated December 15, 2003. The licensee reanalyzed the KNPP Updated Safety Analysis Report (USAR) Chapter 14 non-LOCA events at stretch power uprate (1772 MWt) conditions, and in accordance with approved KNPP reload safety evaluation methods (Letter from K. H. Weinhaue, Nuclear Management Company, to USNRC, "Wisconsin Public Service Corporation Reload Safety Evaluation Methods topical Report," WPSRSEM-NP, Revision 3, dated October 12, 2000, and Letter from J. Lamb, USNRC, to M. Reddemann, Nuclear Management Company, "Kewaunee Nuclear Power Plant - Review for Kewaunee Reload Safety Methods Topical Report WPSRSEM-NP, Revision 3 (TAC-NO. MB0306)," dated September 10, 2001). The licensee performed the Framatome ANP fuel DNBR analysis using the VIPRE subchannel code in accordance with the methodology described in the letter from K. H. Weinhaue, NMC, to USNRC, "Wisconsin Public Service Corporation Reload Safety Evaluation Methods topical Report," WPSRSEM-NP, Revision 3, dated October 12, 2000, and letter from J. Lamb, USNRC, to M. Reddemann, NMC, "Kewaunee Nuclear Power Plant - Review for Kewaunee Reload Safety Methods Topical Report WPSRSEM-NP, Revision 3 (TAC NO. MB0306)," dated September 10, 2001. The

results of the licensee's analysis demonstrate that the DNBR acceptance criteria for this event is satisfied. Based on the results of the licensee's analysis, the NRC staff finds that the acceptance criteria for this event are satisfied considering the use of Framatome ANP fuel under stretch power uprate conditions.

The NRC staff has reviewed the licensee's analyses of the excessive heat removal due to feedwater temperature reduction or flow increase and concludes that the licensee's analyses have adequately accounted for the changes in core design required for operation of the plant at the proposed stretch uprate power level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the SAFDLs are not exceeded. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, 20, and 26 following implementation of the proposed power uprate. Therefore, the NRC staff finds the proposed stretch power uprate acceptable with respect to the excessive heat removal due to feedwater temperature reduction or flow increase.

3.2.2.12.2.2 Excessive Load Increase Incident

The regulatory requirements for this analysis are the same as for the excess heat removal due to feedwater system malfunction event discussed above. This is a cooldown transient and therefore, DNBR is the primary concern for this event. The licensee evaluated events resulting in a rapid increase in SG steam flow that cause a power mismatch between the reactor core power and the SG load demand. Any loading rate in excess of a 10 percent step load increase or a 5 percent per minute ramp load increase in the range of 15 to 95 percent of full power may cause a reactor trip actuated by the RPS. The effect of this transient on the minimum DNBR was evaluated by applying conservatively large deviations on the initial conditions for power, average coolant temperature, and pressurizer pressure at the normal full power operating conditions in order to generate a limiting set of statepoints. These deviations bound the variations that could occur as a result of this event and are applied in the direction that the most adverse impact on DNBR. The reactor condition statepoints were then compared to the conditions corresponding to operation at the DNB safety analysis limit.

The NRC staff previously reviewed and approved the licensee's analyses related to the excessive load increase incident for a Westinghouse 422V+ core at the stretch power uprate level of 1772 MWt as part of KNPP fuel transition amendment. The licensee utilized NRC approved methodologies to evaluate this transient and demonstrated that all acceptance criteria are satisfied. The NRC staff's review and approval of this transient for the Westinghouse 422V+ core is discussed in detail in Section 2.4.2 of the fuel transition amendment.

To support the stretch power uprate LAR, the NRC staff requested that the licensee submit the Framatome ANP fuel non-LOCA transient and accident analysis results. The licensee provided this information in the letter dated December 15, 2003. The licensee reanalyzed the KNPP USAR Chapter 14 non-LOCA events at stretch power uprate (1772 MWt) conditions, and in accordance with approved KNPP reload SE methods. The licensee performed the Framatome ANP fuel DNBR analysis using the VIPRE subchannel code in accordance with the methodology described in the letter from K. H. Weinhaue, NMC, to USNRC, "Wisconsin Public Service Corporation Reload Safety Evaluation Methods topical Report," WPSRSEM-NP, Revision 3, dated October 12, 2000, and letter from J. Lamb, USNRC, to M. Reddemann, NMC,

“Kewaunee Nuclear Power Plant - Review for Kewaunee Reload Safety Methods Topical Report WPSRSEM-NP, Revision 3 (TAC NO. MB0306),” dated September 10, 2001. The results of the licensee’s analysis demonstrate that the DNBR acceptance criteria for this event is satisfied. Based on the results of the licensee’s analysis, the NRC staff finds that the acceptance criteria for this event are satisfied considering the use of Framatome ANP fuel under stretch power uprate conditions.

The NRC staff has reviewed the licensee’s analyses of the excessive load increase incident and concludes that the licensee’s analyses have adequately accounted for the changes in core design required for operation of the plant at the proposed stretch uprate power level. The NRC staff also concludes that the licensee’s analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the SAFDLs are not exceeded. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, 20, and 26 following implementation of the proposed stretch power uprate. Therefore, the NRC staff finds the proposed stretch power uprate acceptable with respect to the excessive load increase incident.

3.2.2.12.2.3 Steamline Break

The steam release resulting from a rupture of a main steam pipe will result in an increase in steam flow, a reduction of coolant temperature and pressure, and an increase in core reactivity. The core reactivity increase may cause a power level increase and a decrease in shutdown margin. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers postulated initial core and reactor conditions; methods of thermal and hydraulic analyses; postulated sequence of events; assumed responses of the reactor coolant and auxiliary systems; functional and operational characteristics of the RPS; required operator actions; core power excursion due to power demand created by excessive steam flow; variables influencing neutronics; and the results of the transient analyses. Acceptance criteria are based on GDC 17 as it relates to the requirement that an onsite and offsite electric power system be provided to permit the functioning of structures, systems, and components (SSCs) important to safety; GDC 27 and GDC 28 as they relate to the RCS being designed with appropriate margin to ensure that the capability to cool the core is maintained; GDC 31 as it relates to the RCS being designed with sufficient margin to ensure that the boundary behaves in a nonbrittle manner and that the probability of propagating fracture is minimized; and GDC 35 as it relates to the RCS and associated auxiliaries being designed to provide abundant emergency core cooling. Specific review criteria are contained in SRP Section 15.1.5.

The NRC staff previously reviewed and approved the licensee’s analyses related to the steamline break for a Westinghouse 422V+ core at the stretch power uprate level of 1772 MWt as part of KNPP fuel transition amendment. The licensee utilized NRC-approved methodologies to evaluate this transient and demonstrated that all acceptance criteria are satisfied. The NRC staff’s review and approval of this transient for the Westinghouse 422V+ core is discussed in detail in Section 2.4.2 of the fuel transition amendment.

To support the stretch power uprate LAR, the NRC staff requested that the licensee submit the Framatome ANP fuel non-LOCA transient and accident analysis results. The licensee provided this information in the letter dated December 15, 2003. The licensee reanalyzed the KNPP USAR Chapter 14 non-LOCA events at stretch power uprate (1772 MWt) conditions, and in

accordance with approved KNPP reload SE methods. The licensee performed the Framatome ANP fuel DNBR analysis using the VIPRE subchannel code in accordance with the methodology described in the letter from K. H. Weinhaue, NMC, to USNRC, “Wisconsin Public Service Corporation Reload Safety Evaluation Methods topical Report,” WPSRSEM-NP, Revision 3, dated October 12, 2000, and letter from J. Lamb, USNRC, to M. Reddemann, NMC, “Kewaunee Nuclear Power Plant - Review for Kewaunee Reload Safety Methods Topical Report WPSRSEM-NP, Revision 3 (TAC NO. MB0306),” dated September 10, 2001. The results of the licensee’s analysis demonstrate that the DNBR acceptance criteria for this event is satisfied. Based on the results of the licensee’s analysis, the NRC staff finds that the acceptance criteria for this event are satisfied considering the use of Framatome ANP fuel under stretch power uprate conditions.

The NRC staff has reviewed the licensee’s analyses of the excessive heat removal due to steamline break and concludes that the licensee’s analyses have adequately accounted for the changes in core design required for operation of the plant at the proposed stretch power uprate level. The NRC staff also concludes that the licensee’s analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the fuel design limits are not exceeded. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 17, 27, 28,31,and 35 following implementation of the proposed stretch power uprate. Therefore, the NRC staff finds the proposed stretch power uprate acceptable with respect to the steamline break.

3.2.2.12.2.4 Loss of External Electric Load

A number of initiating events which are expected to occur with moderate frequency result in unplanned decreases in heat removal by the secondary system. These events result in a sudden reduction in steam flow and, consequently, pressurization events. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff’s review covers the sequence of events, the analytical models used for analyses, the values of parameters used in the analytical models, and the results of the transient analyses. Acceptance criteria are based on GDC 10 as it relates to the RCS being designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; GDC 15 as it relates to the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operations, including AOOs; GDC 17 as it relates to providing onsite and offsite electric power systems to ensure that SSCs important to safety will function during normal operation, including AOOs; and GDC 26 as it relates to the reliable control of reactivity changes to ensure that SAFDLs are not exceeded during normal operation, including AOOs. Specific review criteria are contained in SRP Section 15.2.1-5.

The NRC staff previously reviewed and approved the licensee’s analyses related to the loss of external electric load for a Westinghouse 422V+ core at the stretch power uprate level of 1772 MWt as part of KNPP fuel transition amendment. The licensee utilized NRC-approved methodologies to evaluate this transient and demonstrated that all acceptance criteria are satisfied. The NRC staff’s review and approval of this transient for the Westinghouse 422V+ core is discussed in detail in Section 2.4.2 of the fuel transition amendment.

To support the stretch power uprate LAR, the NRC staff requested that the licensee submit the Framatome ANP fuel non-LOCA transient and accident analysis results. The licensee provided this information in the letter dated December 15, 2003. The licensee reanalyzed the KNPP USAR Chapter 14 non-LOCA events at stretch power uprate (1772 MWt) conditions, and in accordance with approved KNPP reload safety evaluation methods. The licensee performed the Framatome ANP fuel DNBR analysis using the VIPRE subchannel code in accordance with the methodology described in the letter from K. H. Weinhaue, NMC, to USNRC, "Wisconsin Public Service Corporation Reload Safety Evaluation Methods topical Report," WPSRSEM-NP, Revision 3, dated October 12, 2000, and letter from J. Lamb, USNRC, to M. Reddemann, NMC, "Kewaunee Nuclear Power Plant - Review for Kewaunee Reload Safety Methods Topical Report WPSRSEM-NP, Revision 3 (TAC NO. MB0306)," dated September 10, 2001. The results of the licensee's analysis demonstrate that the DNBR acceptance criteria for this event is satisfied. Based on the results of the licensee's analysis, the NRC staff finds that the acceptance criteria for this event are satisfied considering the use of Framatome ANP fuel under stretch power uprate conditions.

The NRC staff has reviewed the licensee's analyses of the loss of external electric load and concludes that the licensee's analyses have adequately accounted for the changes in core design required for operation of the plant at the proposed stretch power uprate level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the SAFDLs and the peak primary and secondary system pressures are not exceeded. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, 17, and 26 following implementation of the proposed stretch power uprate. Therefore, the NRC staff finds the proposed stretch power uprate acceptable with respect to the loss of external electric load.

3.2.2.12.2.5 Loss of Alternating Current (AC) Power to the Plant Auxiliaries

The loss of nonemergency AC power is assumed to result in the loss of all power to the station auxiliaries and the simultaneous tripping of all RCPs. This causes a flow coastdown as well as a decrease in heat removal by the secondary system, a turbine trip, an increase in pressure and temperature of the coolant, and a reactor trip. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers the sequence of events, the analytical model used for analyses, the values of parameters used in the analytical model, and the results of the transient analyses. Acceptance criteria are based on GDC 10 as it relates to the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operation including AOOs; GDC 15 as it relates to the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operation including AOOs; and GDC 26 as it relates to the reliable control of reactivity changes to assure ensure that SAFDLs are not exceeded during normal operation, including AOOs. Specific review criteria are contained in SRP Section 15.2.6.

The NRC staff previously reviewed and approved the licensee's analyses related to the loss of AC power to the plant auxiliaries for a Westinghouse 422V+ core at the stretch power uprate level of 1772 MWt as part of KNPP fuel transition amendment. The licensee utilized NRC approved methodologies to evaluate this transient and demonstrated that all acceptance criteria

are satisfied. The NRC staff's review and approval of this transient for the Westinghouse 422V+ core is discussed in detail in Section 2.4.2 of the fuel transition amendment.

The NRC staff has reviewed the licensee's analyses of the loss of AC power to plant auxiliaries and concludes that the licensee's analyses have adequately accounted for the changes in core design required for operation of the plant at the proposed stretch uprate power level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the peak primary and secondary system pressures are not exceeded. Since this is a heatup transient, DNB is not challenged during this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed stretch power uprate. Therefore, the NRC staff finds the proposed stretch power uprate acceptable with respect to the loss of AC power to the plant auxiliaries.

3.2.2.12.2.6 Loss of Normal Feedwater

A loss of normal feedwater flow could occur from pump failures, valve malfunctions, or a loss of offsite power. Loss of feedwater flow results in an increase in reactor coolant temperature and pressure, which eventually requires a reactor trip to prevent fuel damage. Decay heat must be transferred from fuel following a loss of normal feedwater flow. Reactor protection and safety systems are actuated to provide this function and mitigate other aspects of the transient. The NRC staff's review covers (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC 10 for the RCS being designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs, (2) GDC 15 for the RCS and associated auxiliaries being designed with appropriate margin to ensure that the RCPB will not be breached during normal operations, including AOOs, and (3) GDC 26 for the reliable control of reactivity changes to ensure that SAFDLs are not exceeded during normal operations, including AOOs. Specific review criteria are contained in SRP Section 15.2.7.

In support of its proposed stretch power uprate, the licensee performed a loss of normal feedwater analysis at the uprated power level. The analysis was performed using methodology consistent with the current analysis of record. The NSSS design parameters for the uprated core power of 1772 MWt are used in this analysis. Following a loss of feedwater event, a reactor trip is initiated on either low-low SG water level or steam flow-feedwater flow mismatch coincident with low SG water level. The analysis assumes that the AFW flow from two motor driven AFW pumps is initiated 800 seconds after the reactor trip. The AFW flow will provide sufficient heat sink to prevent excessive heatup or overpressurization of the RCS. The results of the licensee's analysis show that the calculated peak main steam system pressure is less than 110 percent of the SG design pressure, the RCS overpressurization limit is not challenged, and the pressurizer is not filled solid during this transient scenario. The NRC staff noted that the pressurizer sprays and power operated relief valves are assumed to be operable in this transient so as to maximize the potential for pressurizer filling. This will demonstrate that the potential liquid relief from the pressurizer safety valves could not occur. Since this event is bounded by the loss of external electrical load with respect to peak primary and secondary system pressures, the NRC staff find the results of the licensee's analysis acceptable.

The NRC staff has reviewed the licensee’s analyses of the loss of normal feedwater flow event and concludes that the licensee’s analyses have adequately accounted for operation of the plant at the proposed stretch power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of the loss of normal feedwater flow. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed stretch power uprate. Therefore, the NRC staff finds the proposed stretch power uprate acceptable with respect to the loss of normal feedwater flow event.

3.2.2.12.2.7 Loss of Reactor Coolant Flow

A decrease in reactor coolant flow occurring while the plant is at power could result in a degradation of core heat transfer. An increase in fuel temperature and accompanying fuel damage could then result if SAFDL are exceeded during the transient. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers the postulated initial core and reactor conditions, the methods of thermal and hydraulic analysis, the postulated sequence of events, assumed reactions of reactor systems components, the functional and operational characteristics of the RPS, required operator actions, and the results of the transient analyses. Acceptance criteria are based on GDC 10 as it relates to the RCS being designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; GDC 15 as it relates to the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operations, including AOOs; GDC 17 as it relates to providing onsite and offsite electric power systems to ensure that SSCs important to safety will function during normal operation, including AOOs and GDC 26 as it relates to the reliable control of reactivity changes to ensure that SAFDL are not exceeded during normal operation, including AOOs. Specific review criteria are contained in SRP Section 15.3.1/2.

The NRC staff previously reviewed and approved the licensee’s analyses related to the loss of reactor coolant flow for a Westinghouse 422V+ core at the stretch power uprate level of 1772 MWt as part of KNPP fuel transition amendment. The licensee utilized NRC-approved methodologies to evaluate this transient and demonstrated that all acceptance criteria are satisfied. The NRC staff’s review and approval of this transient for the Westinghouse 422V+ core is discussed in detail in Section 2.4.2 of the fuel transition amendment.

To support the stretch power uprate LAR, the NRC staff requested that the licensee submit the Framatome ANP fuel non-LOCA transient and accident analysis results. The licensee provided this information in the letter dated December 15, 2003. The licensee reanalyzed the KNPP USAR Chapter 14 non-LOCA events at stretch power uprate (1772 MWt) conditions, and in accordance with approved KNPP reload safety evaluation methods. The licensee performed the Framatome ANP fuel DNBR analysis using the VIPRE subchannel code in accordance with the methodology described in the letter from K. H. Weinhaue, NMC, to USNRC, “Wisconsin Public Service Corporation Reload Safety Evaluation Methods topical Report,” WPSRSEM-NP, Revision 3, dated October 12, 2000, and letter from J. Lamb, USNRC, to M. Reddemann, NMC, “Kewaunee Nuclear Power Plant - Review for Kewaunee Reload Safety Methods Topical Report WPSRSEM-NP, Revision 3 (TAC NO. MB0306),” dated September 10, 2001. The results of the licensee’s analysis demonstrate that the DNBR acceptance criteria for this event

is satisfied. Based on the results of the licensee's analysis, the NRC staff finds that the acceptance criteria for this event are satisfied considering the use of Framatome ANP fuel under stretch power uprate conditions.

The NRC staff has reviewed the licensee's analyses of the loss of reactor coolant flow and concludes that the licensee's analyses have adequately accounted for the changes in core design required for operation of the plant at the proposed stretch power uprate level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the SAFDLs and the peak primary and secondary system pressures are not exceeded. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, 17, and 26 following implementation of the proposed stretch power uprate. Therefore, the NRC staff finds the proposed stretch power uprate acceptable with respect to the loss of reactor coolant flow.

3.2.2.12.2.8 Locked Rotor Accident

The events postulated are an instantaneous seizure of the rotor or break of the shaft of a RCP in a PWR. Flow through the affected loop is rapidly reduced, leading to a reactor and turbine trip. The sudden decrease in core coolant flow while the reactor is at power results in a degradation of core heat transfer which could result in fuel damage. The initial rate of reduction of coolant flow is greater for the rotor seizure event. However, the shaft break event permits a greater reverse flow through the affected loop later during the transient and, therefore, results in a lower core flow rate at that time. In either case, reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covers the postulated initial and long-term core and reactor conditions, the methods of thermal and hydraulic analysis, the postulated sequence of events, the assumed reactions of reactor system components, the functional and operational characteristics of the RPS, required operator actions, and the results of the transient analyses. Acceptance criteria are based on GDC 17 as it relates to providing onsite and offsite electric power systems to ensure that SSCs important to safety will function; GDC 27 and GDC 28 as they relate to the RCS being designed with appropriate margin to ensure that the capability to cool the core is maintained; and GDC 31 as it relates to the RCS being designed with sufficient margin to ensure that the boundary behaves in a nonbrittle manner and that the probability of propagating fracture is minimized. Specific review criteria are contained in SRP Section 15.3.3/4. Also, the current KNPP licensing basis includes an acceptance criterion to limit the maximum clad temperature to 2700 of to assure coolable core geometry is maintained.

The NRC staff previously reviewed and approved the licensee's analyses related to the locked rotor accident for a Westinghouse 422V+ core at the stretch power uprate level of 1772 MWt as part of KNPP fuel transition amendment. The licensee utilized NRC-approved methodologies to evaluate this transient and demonstrated that all acceptance criteria are satisfied. The NRC staff's review and approval of this transient for the Westinghouse 422V+ core is discussed in detail in Section 2.4.2 of the fuel transition amendment.

To support the stretch power uprate LAR, the NRC staff requested that the licensee submit the Framatome ANP fuel non-LOCA transient and accident analysis results. The licensee provided this information in the letter dated December 15, 2003. The licensee reanalyzed the KNPP USAR Chapter 14 non-LOCA events at stretch power uprate (1772 MWt) conditions, and in

accordance with approved KNPP reload safety evaluation methods. The licensee performed the Framatome ANP fuel DNBR analysis using the VIPRE subchannel code in accordance with the methodology described in the letter from K. H. Weinhaue, NMC, to USNRC, “Wisconsin Public Service Corporation Reload Safety Evaluation Methods topical Report,” WPSRSEM-NP, Revision 3, dated October 12, 2000, and letter from J. Lamb, USNRC, to M. Reddemann, NMC, “Kewaunee Nuclear Power Plant - Review for Kewaunee Reload Safety Methods Topical Report WPSRSEM-NP, Revision 3 (TAC NO. MB0306),” dated September 10, 2001. The results of the licensee’s analysis demonstrate that the DNBR acceptance criteria for this event is satisfied. Based on the results of the licensee’s analysis, the NRC staff finds that the acceptance criteria for this event are satisfied considering the use of Framatome ANP fuel under stretch power uprate conditions.

The NRC staff has reviewed the licensee’s analyses of the locked rotor accident and concludes that the licensee’s analyses have adequately accounted for the changes in core design required for operation of the plant at the proposed stretch uprate power level. The NRC staff also concludes that the licensee’s analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the limits of the fuel clad temperature and the peak primary and secondary system pressures are not exceeded. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 17, 27, 28, and 31 as well as the maximum fuel clad temperature of 2700 of following implementation of the proposed stretch power uprate. Therefore, the NRC staff finds the proposed stretch power uprate acceptable with respect to the locked rotor accident.

3.2.2.12.2.9 Startup of an Inactive Reactor Coolant Loop

Starting the idle RCP without first bringing the hot-leg temperature of the inactive loop close to the core inlet temperature would result in injection of cold water into the core. This injection of cold water into the core could cause a reactivity insertion, and subsequently a power increase due to the effects of moderator density reactivity feedback. The NRC staff’s review covers the sequence of events, the analytical model, the values of parameters used in the analytical model, and the results of the transient analyses. Acceptance criteria are based on GDC 10 and GDC 20 as they relate to the RCS being designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; GDC 15 and GDC 28 as they relate to the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operations, including AOOs; and GDC 26 as it relates to the reliable control of reactivity changes to ensure that SAFDLs are not exceeded during normal operations, including AOOs. Specific review criteria are contained in SRP Section 15.4.4/5

The KNPP TSs limit the reactor power to less than 2 percent rated thermal power when only one RCP is in operation. At this power level, the hot-leg temperature of the inactive loop would already be very close to the core inlet temperature. For this reason, the licensee has determined that no analysis is needed to show that the DNBR limit is satisfied for this event at KNPP. As part of KNPP fuel transition amendment, the NRC staff has previously reviewed and approved this licensee’s assessment and concludes that the KNPP TS will prevent unacceptable results from a potential transient due to startup of an inactive reactor coolant loop. Therefore, an analysis of this event is unnecessary.

3.2.2.12.2.10 Uncontrolled RCCA Withdrawal from Subcritical Condition

An uncontrolled RCCA withdrawal from subcritical condition may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covers (1) the description of the causes of the transient and the transient itself, (2) the initial conditions, (3) the reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the transient analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-20 for the RPS being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems to ensure that SAFDLs are not exceeded during normal operation, including AOOs; and (3) GDC-25 for the functional design of redundant reactivity control systems to assure that SAFDLs are not exceeded in the event of a single malfunction of the reactivity control systems. Specific review criteria are contained in SRP Section 15.4.1.

The NRC staff previously reviewed and approved the licensee's analyses related to the uncontrolled RCCA withdrawal from subcritical condition for a Westinghouse 422V+ core at the stretch power uprate level of 1772 MWt as part of KNPP fuel transition amendment. The licensee incorporated NRC approved methodologies to evaluate this transient and demonstrated that all acceptance criteria are satisfied. The NRC staff's review and approval of this transient for the Westinghouse 422V+ core is discussed in detail in Section 2.4.2.1 of the fuel transition amendment.

To support the stretch power uprate LAR, the NRC staff requested that the licensee submit the Framatome ANP fuel non-LOCA transient and accident analysis results. The licensee provided this information in the letter dated December 15, 2003. The licensee reanalyzed the KNPP USAR Chapter 14 non-LOCA events at stretch power uprate (1772 MWt) conditions, and in accordance with approved KNPP reload safety evaluation methods. The licensee performed the Framatome ANP fuel DNBR analysis using the VIPRE subchannel code in accordance with the methodology described in the letter from K. H. Weinhaue, NMC, to USNRC, "Wisconsin Public Service Corporation Reload Safety Evaluation Methods topical Report," WPSRSEM-NP, Revision 3, dated October 12, 2000, and letter from J. Lamb, USNRC, to M. Reddemann, NMC, "Kewaunee Nuclear Power Plant - Review for Kewaunee Reload Safety Methods Topical Report WPSRSEM-NP, Revision 3 (TAC NO. MB0306)," dated September 10, 2001. The results of the licensee's analysis demonstrate that the DNBR acceptance criteria for this event is satisfied. Based on the results of the licensee's analysis, the NRC staff finds that the acceptance criteria for this event are satisfied considering the use of Framatome ANP fuel under stretch power uprate conditions.

The NRC staff has reviewed the licensee's analyses of the uncontrolled RCCA withdrawal from a subcritical condition and concludes that the licensee's analyses have adequately accounted for the changes in core design required for operation of the plant at the proposed stretch uprate power level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the SAFDLs are not exceeded. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 20, and 25 following implementation of the fuel upgrade and proposed stretch power uprate. Therefore, the NRC staff finds the proposed stretch power

uprate acceptable with respect to the uncontrolled RCCA withdrawal from a subcritical condition.

3.2.2.12.2.11 Uncontrolled RCCA Withdrawal at Power

An uncontrolled RCCA withdrawal at power event may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covers (1) the description of the causes of the AOOs and the description of the event itself, (2) the initial conditions, (3) the reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the associated analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-20 as it relates the RPS being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded during normal operation, including AOOs; and (3) GDC-25 for the functional design of redundant reactivity systems to assure that SAFDLs are not exceeded in the event of a single malfunction of the reactivity control systems. Specific review criteria are contained in SRP Section 15.4.2.

The NRC staff previously reviewed and approved the licensee's analyses related to the uncontrolled RCCA withdrawal at power condition for a Westinghouse 422V+ core at the stretch power uprate level of 1772 MWt as part of KNPP fuel transition amendment. The licensee incorporated NRC-approved methodologies to evaluate this transient and demonstrated that all acceptance criteria are satisfied. The NRC staff's review and approval of this transient for the Westinghouse 422V+ core is discussed in detail in Section 2.4.2.2 of fuel transition amendment.

To support the stretch power uprate LAR, the NRC staff requested that the licensee submit the Framatome ANP fuel non-LOCA transient and accident analysis results. The licensee provided this information in the letter dated December 15, 2003. The licensee reanalyzed the KNPP USAR Chapter 14 non-LOCA events at stretch power uprate (1772 MWt) conditions, and in accordance with approved KNPP reload SE methods. The licensee performed the Framatome ANP fuel DNBR analysis using the VIPRE subchannel code in accordance with the methodology described in the letter from K. H. Weinhaue, NMC, to USNRC, "Wisconsin Public Service Corporation Reload Safety Evaluation Methods topical Report," WPSRSEM-NP, Revision 3, dated October 12, 2000, and letter from J. Lamb, USNRC, to M. Reddemann, NMC, "Kewaunee Nuclear Power Plant - Review for Kewaunee Reload Safety Methods Topical Report WPSRSEM-NP, Revision 3 (TAC NO. MB0306)," dated September 10, 2001. The results of the licensee's analysis demonstrate that the DNBR acceptance criteria for this event is satisfied. Based on the results of the licensee's analysis, the NRC staff finds that the acceptance criteria for this event are satisfied considering the use of Framatome ANP fuel under stretch power uprate conditions.

The NRC staff has reviewed the licensee's analyses of the uncontrolled RCCA withdrawal at power and concludes that the licensee's analyses have adequately accounted for the changes in core design required for operation of the plant at the proposed stretch power uprate level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the SAFDLs are not exceeded.

Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 20, and 25 following implementation of the fuel upgrade and proposed stretch power uprate. Therefore, the NRC staff finds the proposed stretch power uprate acceptable with respect to the uncontrolled RCCA withdrawal at power.

3.2.2.12.2.12 RCCA Misalignment

The NRC staff's review covers the types of control rod misoperations that are assumed to occur, including those caused by a system malfunction or operator error. The review covers (1) descriptions of rod position, flux, pressure, and temperature indication systems, and those actions initiated by these systems (e.g., turbine runback, rod withdrawal prohibit, rod block) which can mitigate the effects or prevent the occurrence of various misoperations; (2) the sequence of events; (3) the analytical model used for analyses; (4) important inputs to the calculations; and (5) the results of the analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the RCS being designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; (2) GDC-20 for the RPS being designed to automatically initiate appropriate systems to ensure that SAFDLs are not exceeded as a result of AOOs; and (3) GDC-25 for the functional design of redundant reactivity systems to assure that SAFDLs are not exceeded in the event of a single malfunction of the reactivity control systems. Specific review criteria are contained in SRP Section 15.4.3.

The NRC staff previously reviewed and approved the licensee's analyses related to the RCCA misalignment transient for a Westinghouse 422V+ core at the stretch power uprate level of 1772 MWt as part of KNPP fuel transition amendment. The licensee utilized NRC-approved methodologies to evaluate this transient and demonstrated that all acceptance criteria are satisfied. The NRC staff's review and approval of this transient for the Westinghouse 422V+ core is discussed in detail in Section 2.4.2.3 of the fuel transition amendment.

To support the stretch power uprate LAR, the NRC staff requested that the licensee submit the Framatome ANP fuel non-LOCA transient and accident analysis results. The licensee provided this information in the letter dated December 15, 2003. The licensee reanalyzed the KNPP USAR Chapter 14 non-LOCA events at stretch power uprate (1772 MWt) conditions, and in accordance with approved KNPP reload safety evaluation methods. The licensee performed the Framatome ANP fuel DNBR analysis using the VIPRE subchannel code in accordance with the methodology described in the letter from K. H. Weinhaue, NMC, to USNRC, "Wisconsin Public Service Corporation Reload Safety Evaluation Methods topical Report," WPSRSEM-NP, Revision 3, dated October 12, 2000, and letter from J. Lamb, USNRC, to M. Reddemann, NMC, "Kewaunee Nuclear Power Plant - Review for Kewaunee Reload Safety Methods Topical Report WPSRSEM-NP, Revision 3 (TAC NO. MB0306)," dated September 10, 2001. The results of the licensee's analysis demonstrate that the DNBR acceptance criteria for this event is satisfied. Based on the results of the licensee's analysis, the NRC staff finds that the acceptance criteria for this event are satisfied considering the use of Framatome ANP fuel under stretch power uprate conditions.

The NRC staff has reviewed the licensee's analyses of the RCCA misalignment transients and concludes that the licensee's analyses have adequately accounted for the changes in core design required for operation of the plant at the proposed stretch power uprate level. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor

protection and safety systems will continue to ensure the SAFDLs are not exceeded. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 20, and 25 following implementation of the proposed stretch power uprate. Therefore, the NRC staff finds the proposed stretch power uprate acceptable with respect to the RCCA misalignment transients.

3.2.2.12.2.13 Chemical and Volume Control System Malfunction

Unborated water can be added to the reactor coolant system, via the chemical and volume control system (CVCS). This may happen inadvertently because of operator error or CVCS malfunction, and cause an unwanted increase in reactivity and a decrease in shutdown margin. The operator must stop this unplanned dilution before the shutdown margin is eliminated. The NRC staff's review covers (1) conditions at the time of the unplanned dilution, (2) causes, (3) initiating events, (4) the sequence of events, (5) the analytical model used for analyses, (6) the values of parameters used in the analytical model, and (7) results of the analyses. The NRC's acceptance criteria are based on (1) GDC-10 for the reactor core and associated coolant, control, and protection systems being designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including AOOs; (2) GDC-15 for the RCS and associated auxiliary, control, and protection systems being designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs; and (3) GDC-26 for the control rods being capable of reliably controlling reactivity changes to assure that SAFDLs are not exceeded during normal operation, including AOOs. Specific review criteria are contained in SRP Section 15.4.6.

The NRC staff previously reviewed and approved the licensee's analyses related to the CVCS malfunction at the stretch power uprate level of 1772 MWt as part of KNPP fuel transition amendment. The licensee utilized NRC-approved methodologies to evaluate this transient and demonstrated that all acceptance criteria are satisfied. The licensees analyses accounted for transition cores through an all Westinghouse 422V+ core. The NRC staff's review and approval of this transient for the stretch power uprate is discussed in detail in Section 2.4.2.4 of fuel transition amendment.

The NRC staff has reviewed the licensee's analyses of the CVCS malfunction transient and concludes that the licensee's analyses have adequately accounted for the changes in core design required for operation of the plant with the Westinghouse 422V+ fuel and at the proposed stretch power uprate. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the SAFDLs are not exceeded. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 20, and 25 following implementation of the fuel upgrade and proposed stretch power uprate. Therefore, the NRC staff finds the proposed fuel upgrade and stretch power uprate acceptable with respect to the CVCS malfunction transient.

3.2.2.12.2.14 Rupture of a Control Rod Drive Mechanism (CRDM) Housing (RCCA Ejection)

Control rod ejection accidents cause a rapid positive reactivity insertion together with an adverse core power distribution, which could lead to localized fuel rod damage. The NRC staff

evaluates the consequences of a control rod ejection accident to determine the potential damage caused to the RCPB and to determine whether the fuel damage resulting from such an accident could impair cooling water flow. The NRC staff's review covers initial conditions, rod patterns and worths, scram worth as a function of time, reactivity coefficients, the analytical model used for analyses, core parameters which affect the peak reactor pressure or the probability of fuel rod failure, and the results of the transient analyses. The NRC's acceptance criteria are based on GDC-28 for ensuring that the effects of postulated reactivity accidents do not result in damage to the RCPB greater than limited local yielding and do not cause sufficient damage to significantly impair the capability to cool the core. Specific review criteria contained in SRP Section 15.4.8 and used to evaluate this accident include: (1) Reactivity excursions should not result in a radially averaged enthalpy greater than 280 cal/gm at any axial location in any fuel rod, and (2) The maximum reactor pressure during any portion of the assumed excursion should be less than the value that will cause stresses to exceed the "Service Limit C" as defined in the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code*.

The NRC staff previously reviewed and approved the licensee's analyses related to the RCCA ejection at the stretch power uprate level of 1772 MWt as part of KNPP fuel transition amendment. The licensee utilized NRC-approved methodologies to evaluate this transient and demonstrated that all acceptance criteria are satisfied. The licensee analyzed both the Framatome ANP fuel and the new Westinghouse 422V+ fuel for this event, and the results demonstrate that the new Westinghouse 422V+ fuel is more limiting. The NRC staff's review and approval of this transient for the stretch power uprate is discussed in detail in Section 2.4.2.15 of the fuel transition amendment.

The NRC staff has reviewed the licensee's analyses of the rod ejection accident and concludes that the licensee's analyses have adequately accounted for operation of the plant for the fuel upgrade and the proposed stretch power uprate and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that appropriate reactor protection and safety systems will prevent postulated reactivity accidents that could (1) result in damage to the RCPB greater than limited local yielding, or (2) cause sufficient damage that would significantly impair the capability to cool the core. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDC-28 following implementation of the proposed fuel upgrade and stretch power uprate. Therefore, the NRC staff finds the proposed fuel upgrade and stretch power uprate acceptable with respect to the rod ejection accident.

3.2.2.12.2.15 Steam Generator Tube Rupture (SGTR)

A SGTR event causes direct release of radioactive material contained in the primary coolant to the environment through the ruptured SG tube and SG safety or atmospheric relief valves. Reactor protection and ESFs are actuated to mitigate the accident and restrict the offsite dose within the guidelines of the 10 CFR 100 limits. The NRC staff's review covers postulated initial core and plant conditions, method of thermal and hydraulic analysis, sequence of events assuming with and without offsite power available, assumed reactions of reactor system components, functional and operational characteristics of the RPS, required operator actions consistent with the plant EOPs, and the results of the accident analysis. A single-failure of mitigating system is assumed for this event. The NRC staff review for SGTR discussed in this section is focused on the thermal and hydraulic analysis for the SGTR in order to: (1) support

the review related to 10 CFR Part 100 for radiological consequences which is addressed below in this safety evaluation, and (2) confirm that there is no overfill of the SG during the mitigation of this event which could cause unacceptable radiological consequences or potential failure of the main steam system. Specific review criteria are contained in SRP Section 15.6.3.

In support of its proposed stretch power uprate, the licensee performed a SGTR thermal-hydraulic analysis for calculation of the radiological consequences. The analysis was performed using methodology consistent with the current analysis of record. The NSSS design parameters for the uprated core power of 1772 MWt are used in this analysis. Following a SGTR, a loss of off-site power is assumed to occur concurrent with the reactor trip resulting in the release of steam to the atmosphere via the SG atmospheric relief valves and/or safety valves. Consistent with the current analysis, the licensee assumes that the operators have completed the actions necessary to terminate the equilibrium break flow and the steam releases from the ruptured SG in 30 minutes after the event initiation. The resulting break flow mass transfer is then used to calculate the radiological consequences of the SGTR. In response to the NRC staff question regarding whether or not the 30 minutes operator action time for terminating the break flow is achievable, the licensee indicated that another detailed thermal-hydraulic analyses has been performed at the uprated power of 1772 MWt to evaluate the impact on the radiological consequences of the SGTR break flow continuing longer than 30 minutes and to evaluate the potential for SG overfill. This analysis used operator action times, supported by Kewaunee simulator exercises, leading to SG isolation within 30 minutes and break flow termination at approximately 49 minutes. The calculated radiological consequences are bounded by the results of the analysis assuming a constant break flow of 30 minutes. Also, the results of the analysis indicate that recovery actions can be performed to terminate the primary to secondary break flow before overfill of the ruptured SG would occur. Since the licensing analysis provided a bounding consequences of the event, the NRC staff finds the licensee's SGTR analysis acceptable.

The NRC staff has reviewed the licensee's analysis of the SGTR accident and concludes that the licensee's analysis has adequately accounted for operation of the plant at the proposed stretch power uprate and was performed using acceptable analytical methods and approved computer codes. The NRC staff further concludes that the assumptions used in this analysis are conservative and that the event does not result in an overfill of the SG. Therefore, the NRC staff finds the proposed stretch power uprate acceptable with respect to the SGTR.

3.2.2.12.2.16 Anticipated Transients Without Scram (ATWS)

ATWS is defined as an AGO followed by the failure of the reactor portion of the protection system specified in GDC-20. 10 CFR 50.62 provides the regulations regarding ATWS, and requires that:

- Each PWR must have equipment that is diverse from the reactor trip system to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must perform its function in a reliable manner and be independent from the existing reactor trip system, and
- Each PWR manufactured by Combustion Engineering (CE) or Babcock and Wilcox (B&W) must have a diverse scram system (DSS). This scram system must be designed to perform its function in a reliable manner and be independent from the existing reactor trip system.

The NRC staff's review is conducted to ensure that the above requirements are satisfied and that the setpoints for the ATWS mitigating system actuation circuitry (AMSAC) and DSS remain valid for the proposed stretch power uprate. In addition, for plants where a DSS is not specifically required by 10 CFR 50.62, the NRC staff verifies that the consequences of an ATWS are acceptable. The acceptance criteria is that the peak primary system pressure should not exceed the ASME Service Level C limit of 3200 psig. The peak ATWS pressure is primarily a function of the moderator temperature coefficient (MTC) and the primary system relief capacity. The NRC staff reviews (1) the limiting event determination, (2) the sequence of events, (3) the analytical model and its applicability, (4) the values of parameters used in the analytical model, and (5) the results of the analyses. If the licensee relies upon generic vendor analyses, the NRC staff reviews the licensee's justification of the applicability of those analyses to the plant under review and the operating conditions for the proposed stretch power uprate.

KNPP has satisfied the requirements of 10 CFR 50.62 by installing AMSAC. As a supplement to AMSAC, KNPP has also installed a DSS. KNPP installed a DSS in 1998 in response to AFW pump net positive suction head (NPSH) concerns. Following implementation of the stretch power uprate, KNPP will continue to maintain an operable AMSAC system in compliance with the requirements of the ATWS rule.

KNPP reanalyzed the loss-of-normal feedwater (LONF) ATWS event to demonstrate that all appropriate acceptance criteria are satisfied under the stretch power uprate conditions. These acceptance criteria include primary system pressure less than 3200 psig, and adequate AFW pump NPSH available. The LONF event is the limiting ATWS event with respect to RCS pressure. The analyses were performed assuming 1780 MWt NSSS power and a moderator temperature coefficient (MTC) of -8 pcm/°F. The MTC value is a key input to the ATWS analysis, and this value is consistent with the basis for the ATWS rule and also with KNPP TS 3.1.f.4, which ensures that the reactor will have an MTC no less negative than -8 pcm/°F for 95 percent of the cycle time at full power. The analyses also assumed two pressurizer power operated relief valves available, consistent with the analytical basis for the ATWS rule. The licensee used the LOFTRAN computer code to perform the KNPP specific ATWS analysis for the stretch power uprate, consistent with the analysis basis for the ATWS rule. The licensee did not credit the DSS system in the peak primary system pressure analysis, but does credit the DSS in the AFW NPSH analysis. The peak primary system pressure obtained from this analysis is 2747 psia, which meets the acceptance criteria of less than 3200 psig. The licensee's results also demonstrate that the AFW pump NPSH requirements will be satisfied at the stretch power uprate conditions.

The NRC staff has reviewed the information submitted by the licensee related to ATWS and concludes that the licensee has adequately accounted for the effects of the proposed stretch power uprate on ATWS. The NRC staff concludes that the licensee has demonstrated that the AMSAC will continue to meet the requirements of 10 CFR 50.62 following implementation of the proposed stretch power uprate. The licensee has shown that the plant is not required by 10 CFR 50.62 to have a DSS. Additionally, the licensee has demonstrated, through acceptable analyses, that the peak primary system pressure following an ATWS event will remain below the acceptance limit of 3200 psig. Based on this, the NRC staff concludes that the plant design will continue to meet the requirements of 10 CFR 50.62 following implementation of the proposed stretch power uprate. Therefore, the NRC staff finds the proposed stretch power uprate acceptable with respect to ATWS.

3.2.2.12.2.17 Station Blackout (SBO)

SBO refers to a complete loss of ac power to the essential and nonessential switchgear buses in a nuclear power plant. SBO involves the loss of offsite power concurrent with a turbine trip and failure of the onsite emergency ac power system. SBO does not include the loss of available ac power to buses fed by station batteries through inverters or the loss of power from "alternate ac sources" (AACs). The NRC staff's review focuses on the impact of the proposed stretch power uprate on the plant's ability to cope with and recover from a SBO event for the period of time established in the plant's licensing basis. The NRC's acceptance criteria for SBO are based on 10 CFR 50.63. Specific review criteria are contained in SRP Sections 8.1 and Appendix B to SRP Section 8.2.

The licensee provided the results of its evaluation relative to the stretch power uprate affecting the plant coping capability in a SBO. RCS inventory control is accomplished by the use of a single charging pump powered by an alternate AC power source, drawing suction from the refueling water storage tank (RWST) and discharging to the RCP seals and RCS loop B cold leg. The flow path and valve requirements, including station battery power and nitrogen backup for air operated valves, for accomplishing RCS inventory control are not impacted by stretch power uprate and remain the same as described in the original SBO mitigation strategy. The licensee's evaluation of the system impacted by the stretch power uprate do not identify any changes to design or operating conditions that adversely affect the ability to provide safe shutdown for a SBO initiated from stretch power uprate conditions.

The NRC staff has reviewed the licensee's assessment of the effects of the proposed stretch power uprate on the plant's ability to cope with and recover from a SBO event for the period of time established in the plant's licensing basis. The NRC staff concludes that the licensee has adequately evaluated the effects of the proposed stretch power uprate on SBO and demonstrated that the plant will continue to meet the requirements of 10 CFR 50.63 following implementation of the proposed stretch power uprate. Therefore, the NRC staff finds the proposed stretch power uprate acceptable with respect to SBO.

3.2.3 <u>Conclusion</u>

The NRC staff has reviewed the licensee's safety analyses of the impact of the proposed stretch power uprate on (1) fuel design, (2) nuclear design, (3) thermal-hydraulic design, (4) performance of control and safety systems connected to the NSSS, and (5) LOCA and non-LOCA transient analyses. The NRC staff concludes that the results of the licensee's analyses related to these areas continue to meet the applicable acceptance criteria following implementation of the proposed stretch power uprate. Where additional evaluations/analyses were necessary, the NRC staff has reviewed these evaluations and analyses and finds that the licensee has satisfactorily addressed the areas discussed above, the supporting safety analyses were performed using NRC-approved methods, the input parameters of the analyses adequately represent the plant conditions at the proposed uprated power level, and the analytical results meet the applicable acceptance criteria. Based on the above, the NRC staff finds the proposed 6.0-percent stretch power uprate acceptable with respect reactor systems performance.

3.3 Electrical Systems

3.3.1 Environmental Qualification of Electrical Equipment

3.3.1.1 Regulatory Evaluation

The term environmental qualification (EQ) applies to equipment important-to-safety to assure this equipment remains functional during and following design basis events. The NRC staff's review covers the environmental conditions that could affect the design and safety functions of electrical equipment including instrumentation and control. The NRC staff's review is to ensure compliance with the acceptance criteria thus ensuring that the equipment continues to be capable of performing its design-basis safety functions under all normal environmental conditions, AOOs, and accident and post accident environmental conditions. Acceptance criteria are based on 10 CFR 50.49 as it relates to specific requirements regarding the qualification of electrical equipment important-to-safety that is located in a harsh environment. Specific review criteria are contained in SRP Section 3.11.

3.3.1.2 Technical Evaluation

The proposed power uprate will change the accident/post-accident temperature profiles inside containment impacting all important-to-safety equipment located inside containment. By letters dated December 15, 2003, and January 30, 2004, the licensee stated that all environmentally qualified electrical equipment has been evaluated for the resulting environmental conditions resulting from the 6.0-percent stretch power uprate and will remain environmentally qualified.

3.3.1.3 Conclusion

The NRC staff has reviewed the licensee's submittals of the effects of the proposed power uprate on the EQ of the electrical equipment and concluded that the resulting environmental conditions resulting from the 6.0-percent power uprate will remain environmentally qualified and, therefore, the design is acceptable.

3.3.2 Offsite Power System

3.3.2.1 Regulatory Evaluation

Prior to the introduction of GDC 17 of Appendix A to 10 CFR Part 50, AEC Criterion 39 was used to evaluate the adequacy of the electric power systems. Criterion 39 requires that sufficient offsite and redundant, independent, and testable standby auxiliary sources of electrical power are provided to attain a prompt shutdown and continued maintenance of the plant in a safe condition under all credible circumstances. The capacity of the power sources is adequate to accomplish all required engineered safety features functions under all postulated design-basis accident conditions. Acceptance criteria are based on Criterion 39.

3.3.2.2 Technical Evaluation

3.3.2.2.1 Grid Stability Technical Evaluation

By letter dated November 05, 2003, the licensee provided additional information for the grid stability analysis including assumptions and results for the stretch power uprate condition. American Transmission Company (ATC) performed the grid stability study for the licensee on October 13, 2003, for a gross output of 597 MWe at KNPP. From a gross output of 559 MWe at KNPP, the study evaluated a 38-MWe increase implemented in two phases: (1) a 10-MWe addition in 2003, and (2) the remaining 28-MWe addition in 2004. The first phase has already been completed with the implementation of the 1.4-percent MUR power uprate in July 2003. The second phase is currently planned for implementation in March 2004 after the NRC staff approval for the 6.0-percent stretch power uprate.

ATC determined that several transmission system upgrades are required to address pre-existing stability issues, which will be corrected prior to implementation of the second phase (28 MWe) of the power uprate. These transmission system upgrades or interim solutions will be installed by ATC prior to implementation of the 6.0-percent stretch power uprate. The required installation of transmission system upgrades and possible revision of transmission system operating guides has been captured in the KNPP site corrective action process to be completed prior to implementation of the 6.0-percent stretch power uprate, and will be documented as a prerequisite in the uprate implementation plan.

3.3.2.2.2 Main Generator Technical Evaluation

The main generator is rated at 622.389 megavolts-amperes (MVA), 560.15 MWe at 0.9 power factor (pf). Once the power uprate is implemented, the main generator will operate within its design rating. The NRC staff was potentially concerned about the impact of the stretch power uprate on MVARs supplied by the main generator. By letter dated November 5, 2003, the licensee confirmed that the generator output at the 6.0-percent stretch power uprate conditions is within the existing generator capability curve. The existing generator, exciter and cooling equipment are adequate to support unit operation at the 6.0-percent stretch power uprate conditions. Once the generator is connected to the grid, the nominal output voltage of the generator to the electrical grid is determined by the grid, and can only be varied by the unit generator a very small amount. Therefore, the voltages at the plant are not adversely affected. Prior to the power uprate, the real power supplied by the main generator was 565.15 MWe at 0.9 pf and reactive power was 272 megavolts-amperes reactive (MVAR). At the stretch power uprate conditions, the real output of the main generator will be 595.7 MWe at 0.957 pf and the reactive power output will be 180. 3 MVAR. While connected to the grid, the plant can vary the MVARs supplied to the grid. The grid voltage and power requirements are maintained by the Energy Supply and Control group of ATC, and the grid voltage range maintained will not change after the 6.0-percent stretch power uprate. Since operation will remain within the capability curve, and voltages at the plant will not change, there are no adverse effects on voltage or operation of plant equipment. If the Energy Supply and Control group of ATC requests an increase in MVARs to support grid voltage, the licensee will have to reduce power in order to stay within the generator capability curve. By letter dated December 15, 2003, the licensee provided additional information in support of the requested change to the KNPP

Operating License. KNPP will not exceed the design limits of the main generator since operating procedures require the plant to remain within the main generator capability curve which is based on a design limit of 622.389 MVA.

The NRC staff reviewed the licensee's submittals and concluded that the plant will continue to operate the main generator within its design rating at the 6.0-percent stretch power uprate conditions and, therefore, the design is acceptable.

3.3.2.2.3 Main Transformer Technical Evaluation

The MT is rated at 649.5 MVA. The maximum MVA capability of the main generator is at 622.389 MVA, which is within the rating of the main transformer. However, the NRC staff was concerned that the MT is not capable of supporting station operation at the 6.0-percent stretch power uprate conditions during the operation of the main generator in the leading mode. The maximum amount of reactive power at the MT secondary that can be accepted is limited to 262 MVAR. By letter dated November 5, 2003, the licensee stated that plant procedures would require operation within the generator capability curve. Therefore, maintaining operation within the generator capability curve prevents exceeding the limit of 262 MVARs at the MT.

The NRC staff reviewed the licensee's submittals and concluded that the anticipated power uprate of 6.0-percent is below the maximum main transformer design rating and, therefore, operating the main power transformers at the 6.0-percent stretch power uprate conditions is acceptable.

3.3.2.2.4 Isolated Phase Bus Technical Evaluation

The Isolated Phase (isophase) Bus connects the main generator to the primary windings of the main transformer and the station power transformers. With the power uprate of 6.0-percent, the current in the main section will be 18.9 kiloamperes (kA) and the current in the branch section will be 1,361 amperes (A). This is below the design rating of 20 kA for the main section and 1,600 A for the branch section.

The NRC staff reviewed the licensee's submittals and concluded that the impact of the 6.0-percent stretch power uprate is below the design rating of the isophase bus and, therefore, operating the isophase bus at the 6.0-percent stretch power uprate conditions is acceptable.

3.3.2.2.5 Main Auxiliary Transformer Technical Evaluation

The main auxiliary transformer (MAT) is rated at 24/32/40 MVA with OA/FOA/FOA rating. The term OA/FOA/FOA is defined as oil-immersed self-cooled/forced-air, forced-oil-cooled/forced-air, forced-oil-cooled. The MAT is capable of supporting station operation at the 6.0-percent stretch power uprate conditions with the balance of plant (BOP) loads operating at their power operating points.

The NRC staff reviewed the licensee's submittals and concluded that the MAT will operate at its rated loading and, therefore, operating the MAT at the 6.0-percent stretch power uprate is acceptable.

3.3.2.2.6 Reserve Auxiliary Transformer Technical Evaluation

The reserve auxiliary transformer (RAT) is rated at 24/32/40 MVA with OA/FOA/FOA rating. The RAT is capable of supporting station operation at the 6.0-percent stretch power uprate conditions with the BOP loads operating at their power operating points.

The NRC staff reviewed the licensee’s submittals and concluded that the RAT will operate at its rated loading and, therefore, operating the RAT at the 6.0-percent stretch power uprate is acceptable.

3.3.2.2.7 Medium Voltage Motors Technical Evaluation

The RCPs are rated at 6,000 horsepower (hp). By letter dated November 5, 2003, the licensee provided additional information in support of the requested change to the KNPP Operating License. At the 6.0-percent stretch power uprate conditions, the RCPs will operate at 5942 hp, which is below their design rating.

The Feedwater Pumps (FWP) and the Condensate Pumps are rated at 5,000 hp and 1,500 hp, respectively. With the stretch power uprate, the FWPs and the Condensate Pumps will operate at 5,150 hp (1.03 service factor (SF)) and 1,528 hp (1.02 SF), respectively. The NRC staff was concerned that by operating the motors at the SF, the life of the motor insulation is reduced by 50-percent if the temperature is increased by 8-12 degrees above motor nameplate full load temperature. By letter dated November 5, 2003, the licensee stated that National Electrical Manufacturers Association (NEMA) MG-1 allowed rise for the motor is 85^0 Celsius (C), so the allowed motor stator temperature would be 125^0 C. The FWPs typically run less than 110^0 C motor stator temperature and the condensate pump motors typically run at less than 100^0 C motor stator temperature. The NRC staff was also concerned that MG 1-20.45 states that induction motors shall operate successfully under running conditions at rated loads with a ±10 percent variation of voltage. By letter dated December 15, 2003, the licensee stated that the stretch power uprate will cause a voltage reduction of 0.3 percent in the buses feeding the motors and operation at 3,965 volts, which is within 1 percent of the ideal operating point of 4,000 volts.

The licensee currently performs activities to identify motor capability degradation. The licensee’s Motor Preventive Maintenance (PM) program activities determine a degrading trend in capability prior to failure. During refueling PMs, the insulation resistance is measured to check that it does not drop below established values. The licensee’s engineering staff will be informed if the resistance measured is below the established values. The air inlet filters and lubricating oil are changed, if the motor is so equipped. Also, these motors have temperature monitoring instrumentation which alarms in the control room before the motors exceed their class rise limits for temperature.

The NRC staff was also concerned that the licensee has applied Institute of Electrical and Electronics Engineers Standard 666 (voltage variation) and NEMA MG-1 SF inappropriately, because they cannot be used concurrently. By letter dated January 30, 2004, the licensee stated that due to the stretch power uprate, the condensate pump motor temperature is expected to increase by about 2.3 °C and the feedwater pump motor by about 3.9 °C. The licensee expects the motors to operate at full load for 21.75 years due to the reduction of the insulation life by 25 percent.

The licensee performed a probabilistic risk assessment evaluation on the effect of a 25 percent increase in the feedwater pump motor failure rate, which showed that the change in the core damage frequency (CDF) was less than the 1.0E-6 Delta - CDF criteria for a significant risk impact. KNPP has operated for 29 years and in that time there has been one FWP failure and one condensate pump motor failure.

The NRC staff reviewed the licensee's submittals and concluded that the licensee has provided adequate justification that the subject motors can be operated during expected periods of power operation and, therefore, operating the motors at the 6.0-percent stretch power uprate is acceptable.

3.3.2.2.8 4,160 Volts Alternating Current Switchgear Technical Evaluation

The circuit breakers current interrupting capabilities for the non-safety medium voltage buses 1-1 through 1-4 are less than the calculated possible fault that these breakers are expected to handle. For buses 1-5 and 1-6, the fault currents expected are within the rated current interrupting capabilities of these breakers. The NRC staff was concerned that the breakers may not be capable of interrupting a fault current higher than the maximum calculated fault current. By letters dated November 5, December 15, 2003, and January 30, 2004, the licensee provided additional information regarding WCAP-16040-P NSSS/BOP Licensing Report which discusses 4,160 volts alternating current (VAC) switchgear. It states that there was no new equipment that changed the short circuit characteristics of the installed equipment at the medium voltage level. This statement means that there is no change in the available three phase or asymmetrical fault current at any of the 4,160 volt buses due to the stretch power uprate. It also states that buses 1-1 through 1-4 overduty condition was previously evaluated and found acceptable by NRC letter to the licensee dated June 2, 1992. The licensee also provided the test report performed by McGraw-Edison that showed that these breakers are capable of interrupting the calculated possible fault current for buses 1 through 4. The licensee also provided the maximum available three phase or asymmetrical fault current that is below the ratings of the test report.

The NRC staff concludes that there is no new equipment installed or modifications to the existing equipment that could affect the fault current available at these buses under the proposed 6.0-percent stretch power uprate conditions. Therefore, the 4,160 VAC Switchgear is acceptable.

3.3.2.3 Conclusion

The NRC staff has reviewed the licensee's submittals for the effect of the proposed 6.0-percent stretch power uprate on the offsite power system and concludes that the offsite power system will continue to meet the requirements of Criterion 39 following implementation of the proposed 6.0-percent stretch power uprate. The NRC staff further concludes that the impact of the proposed 6.0-percent stretch power uprate on grid stability is insignificant. Therefore, the NRC staff finds the proposed 6.0-percent stretch power uprate acceptable with respect to the offsite power system.

3.3.3 Emergency Diesel Generators

3.3.3.1 Regulatory Evaluation

The AC onsite power system includes those standby power sources, distribution systems, and auxiliary supporting systems provided to supply power to the safety-related equipment. The NRC staff's review covers the descriptive information, analyses, and referenced documents for the AC onsite power system. Acceptance criteria are based on Criterion 39 as it relates to the capability of the AC onsite power system to perform its intended functions during all plant operating and accident conditions. Specific review criteria are contained in SRP Sections 8.1 and 8.3.1.

3.3.3.2 Technical Evaluation

The emergency diesel generators (EDGs) are designed to furnish reliable ac power for a safe plant shutdown and for operation of engineered safeguards, when no offsite power is available. The engineered safeguard loads have not changed. The capacity of each EDG is adequate to support the operation of required engineered safeguards under design-basis accident conditions.

3.3.3.3 Conclusion

The NRC staff has reviewed the licensee's submittals for the effect of the proposed 6.0-percent stretch power uprate on the ac onsite power system and concludes that the licensee has adequately accounted for the effects of the proposed power uprate on the system's functional design. The NRC staff further concludes that the ac onsite power system will continue to meet the requirements of Criterion 39 following implementation of the proposed 6.0-percent stretch power uprate. Therefore, the NRC staff finds the proposed 6.0-percent stretch power uprate acceptable with respect to the onsite ac power system.

3.3.4 Direct Current (DC) Distribution System

3.3.4.1 Regulatory Evaluation

The dc power systems include those dc power sources and their distribution systems and auxiliary supporting systems provided to supply motive or control power to safety-related equipment. The NRC staff's review covers the information, analyses, and referenced documents for the dc onsite power system. Acceptance criteria are based on Criterion 39 and 10 CFR 50.63 as they relate to the capability of the dc onsite electrical power to facilitate the functioning of SSCs important to safety.

3.3.4.2 Technical Evaluation

The dc distribution system is designed to supply power during normal, shutdown, accident and post-accident conditions. No load additions or modifications are going to be made as a result of the proposed stretch power uprate; therefore, the 6-percent stretch power uprate does not affect the dc system.

3.3.4.3 Conclusion

The NRC staff has reviewed the licensee's submittals for the effect of the proposed stretch power uprate on the dc onsite power system and concludes that the 6-percent stretch power uprate does not affect the dc system. The NRC staff further concludes that the dc onsite power system will continue to meet the requirements of Criterion 39 following implementation of the proposed 6.0-percent stretch power uprate. Therefore, the NRC staff finds the proposed 6.0-percent stretch power uprate acceptable with respect to the dc onsite power system.

3.3.5 Station Blackout

3.3.5.1 Regulatory Evaluation

SBO refers to the complete loss of ac electric power to the essential and nonessential switchgear buses in a nuclear power plant. SBO involves the loss of offsite power concurrent with turbine trip and failure of the onsite emergency ac power system. SBO does not include the loss of available ac power to buses fed by station batteries through inverters or the loss of power from AAC. The NRC staff's review focuses on the impact of the proposed stretch power uprate on the plant's ability to cope with and recovery from an SBO event as based on 10 CFR 50.63. Specific review criteria are contained in SRP Sections 8.1 and Appendix B to SRP 8.2.

3.3.5.2 Technical Evaluation

The evaluation of a SBO event for the KNPP was performed in accordance with the requirements of RG 1.155, "Station Blackout." This evaluation determined an acceptable SBO duration for KNPP of 4 hours. This 4-hour coping duration was based on the reliability and configuration of the offsite power system and the reliability of the EDGs. To provide assurance that the plant could cope with a SBO of 4 hours duration, several factors were considered. These areas included the following: (1) condensate inventory, (2) Class 1E battery capacity, (3) compressed air, (4) effects of loss of ventilation, (5) containment isolation, and (6) reactor vessel inventory.

The licensee has determined that the only factor potentially affected by the proposed stretch power uprate is the condensate inventory required to provide decay heat removal for the 4-hour duration. The KNPP TS 3.4.c, "Condensate Storage Tank," requires a minimum volume in the condensate storage tank (CST) of 39,000 gallons. According to the information submitted by the licensee, the required CST minimum inventory must be raised from 39,000 gallons to 41,500 gallons in order for the plant to be able to cope with and recover from an SBO event.

The flow path and valve requirements, including station battery power and nitrogen backup supply for air operated valves for accomplishing decay heat removal and RCS inventory controls are not impacted by the stretch power uprate and remain the same as described in the original mitigation strategy.

3.3.5.3 Conclusion

The NRC staff has reviewed the licensee's submittals on the effect of the proposed stretch power uprate on the plant's ability to cope with and recover from an SBO event for the period of time established on the plant's licensing basis. The only plant change that was required as a

result of SBO evaluation and/or re-analysis was an increase in the condensate TS inventory from 39,000 gallons to 41,500 gallons. With this one change, the NRC staff concludes that the licensee has adequately evaluated the effects of the proposed 6.0-percent stretch power uprate on SBO and demonstrated that the plant will continue to meet the requirements of 10 CFR 50.63 following the implementation of the proposed 6.0-percent stretch power uprate. Therefore, the NRC staff finds the proposed 6.0-percent stretch power uprate acceptable with respect to SBO requirements.

3.3.6 Electrical Systems Conclusion

The NRC staff has reviewed the licensee's safety analyses of the impact of the proposed 6.0-percent stretch power uprate on (1) EQ of electrical equipment, (2) grid stability, including performance of the main generator, main transformer, isophase bus, and unit auxiliary transformers, (3) EDG, (4) DC distribution system, and (5) SBO. Results of these evaluations show that the increase in core thermal power would have negligible impact on the grid stability, SBO, or the EQ of electrical components. The NRC staff concludes that the results of the licensee's analyses related to these areas meet the requirements of the AEC criterion as issued in the "Safety Evaluation of the Kewaunee Nuclear Power Plant" on July 24, 1972, supplemented December 18, 1972, and May 10, 1973, 10 CFR 50.63, and 10 CFR 50.49 following the implementation of the proposed 6.0-percent stretch power uprate. Therefore, the NRC staff finds the proposed 6.0-percent stretch power uprate acceptable with respect to electrical systems.

3.4 Civil and Engineering Mechanics

3.4.1 Regulatory Evaluation

The mechanical and civil engineering branch technical evaluation included the structural and functional integrity of piping systems, components and their supports, including core support structures, which are designed in accordance with the rules of the ASME (ASME) *Boiler and Pressure Vessel Code*, Section III, Division 1, USAS B31.1 Power Piping Code, and GDC 1, 2, 4, 10, 14, and 15. The NRC staff review focused on verifying that the licensee has provided reasonable assurance of the structural and functional integrity of piping systems, components, component internals, and their supports under normal and vibratory loadings, including those due to fluid flow, postulated accidents and natural phenomena such as earthquakes.

The acceptance criteria are based on continued conformance with the requirements of the following regulations: (1) 10 CFR Part 50, 50.55a and GDC 1 as they relate to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed, (2) GDC 2 as it relates to structures and components important to safety being designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions, (3) GDC 4 as it relates to structures and components important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions of normal and accident conditions, (4) GDC 10, as it relates to reactor internals, requires that reactor internals shall be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs, (5) GDC 14 as it relates to the RCPB being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture, and (6) GDC 15 as it relates to the

RCS being designed with sufficient margin to ensure that the design conditions are not exceeded.

The specific review areas are contained in the NRC SRP Section 3.9. The review also includes the plant specific provisions of generic letter (GL) 89-10 and GL 96-05, as related to plant specific program for motor-operated valves, GL 95-07, as related to the pressure locking and thermal binding for safety-related gate valves, and the plant-specific evaluation of the GL 96-06 program regarding the over-pressurization of isolated piping segments.

3.4.2 Technical Evaluation

The NRC staff reviewed the KNPP power uprate amendment, as it relates to the effects of the power uprate on the structural and pressure boundary integrity of the NSSS and BOP systems. Affected components in these systems included piping, in-line equipment and pipe supports, the RPV, CSS, reactor vessel internals (RVI), SG, CRDM, RCP, and pressurizer. The NRC staff's SE concerning the effects of the power uprate on the pertinent components is provided below.

3.4.2.1 Reactor Vessel

The proposed power uprate will increase the core power by approximately 6.0 percent above the currently authorized power level of 1673 MWt. The licensee reported that the power increase will result in changing the design parameters given in Table 2.1-1, Attachment 4 of the May 22, 2003, application.

The licensee evaluated the reactor vessel for the effects of the revised design conditions provided in Table 2.1-1 of the May 22, 2003, application with respect to the core power level of 1772 MWt. The evaluation was performed for the limiting vessel locations with regard to stresses and cumulative fatigue usage factors (CUFs) in each of the regions, as identified in the reactor vessel stress reports for the core power uprated conditions. The regions of the reactor vessel affected by the power uprate include outlet and inlet nozzles, the RPV (main closure head flange, studs, and vessel flange), CRDM housing, safety injection nozzles, external supports brackets, bottom head to shell juncture, core support guides, and the instrumentation tubes. In its amendment request, the licensee indicated that the evaluation of the reactor vessel was performed in accordance with the ASME Boiler and Pressure Vessel (B&PV) Code, Section III, 1968 Edition with Addenda through the Winter 1968, which is the code of record. Table 5.1-1 of the May 22, 2003, application provides the calculated maximum stresses and CUFs for the reactor vessel critical locations. The results indicate that the maximum primary plus secondary stresses are within the code allowable limits, and the CUFs remain below the allowable ASME Code limit of 1.0. Therefore, the NRC staff agrees with the licensee's conclusion that the current design of the reactor vessel continues to be in compliance with licensing basis codes for the proposed stretch power uprate condition.

3.4.2.2 Reactor Core Support Structures and Vessel Internals

The licensee evaluated the reactor vessel core support and internal structures. The limiting reactor internal components evaluated include the lower core plate, lower support columns, core barrel, baffle plates, baffle/barrel region bolts, guide tubes and support pins, upper core plate, and upper support columns. The licensee indicated that the reactor internal components

were not licensed to the ASME B&PV Code. However, the design of the KNPP reactor internals was evaluated in accordance with requirements of Subsection NG of the 1989 Edition of the ASME Section III Code.

The licensee evaluated these critical reactor internal components considering the revised design conditions provided in Table 2.1-1 of the May 22, 2003, application for KNPP for the requested power level of 1772 MWt. The licensee indicated that the calculated stress for the limiting reactor internals are acceptable within the Code allowable limits. The calculated CUFs as provided in Table 5.2-1 of the amendment request are less than the ASME code allowable limit of 1.0. In addition, the licensee evaluated the vibration, which was found to remain within the allowable limits for the proposed power uprate condition. Based on the above evaluations, the NRC staff agrees with the licensee's conclusion that the reactor internal components at KNPP will be structurally adequate for the proposed 6.0-percent stretch power uprate.

3.4.2.3 Control Rod Drive Mechanisms

The pressure boundary portion of the CRDMs are those exposed to the vessel/core inlet fluid. KNPP has the L-106A CRDMs, full-length mechanisms manufactured by Westinghouse. The licensee evaluated the adequacy of the CRDMs by reviewing the original E-Specification and the generic evaluation for L-106A CRDMs to compare the design-basis input parameters against the revised design conditions in Table 2.1-1 of Appendix 4 to the May 22, 2003, application for the power uprate. Table 5.4-1 of the May 22, 2003, amendment request indicated that the key input parameters such as the hot-leg maximum temperature and the temperature fluctuation for the uprated power condition are bounded by the design-basis analysis. The power uprate evaluation was performed using Section III of ASME B&PV Code, 1965 Edition with addenda through Summer 1966, which is the Code of record. Tables 5.4-2 and 5.4-3 of Attachment 4 to the May 22, 2003, application provides the calculated stresses and CUFs for the critical CRDM locations at the proposed power uprate conditions, which are less than the ASME Code allowable limits..

On the basis of its review, the NRC staff concurs with the licensee's conclusion that the current design of CRDMs continues to be in compliance with licensing basis codes and standards for the proposed 6.0-percent stretch power uprate.

3.4.2.4 Steam Generators

The licensee reviewed the existing structural and fatigue analyses of the SGs at KNPP and compared the power uprate conditions with the design parameters of the analysis of record for the Model 54F SGs at KNPP. The comparison of key parameters are shown in Table 5.7-1 of Appendix 4 to the May 22, 2003, application for the current rated power and the proposed power uprate conditions.

As a result of its review, the licensee indicated that all components, except the feedwater nozzle, thermal sleeve, and the J-nozzle-to-feed ring weld fatigue analyses, experience the primary or secondary side temperature and pressure gradients when operating at the power uprate condition are bounded by the KNPP existing design-basis analyses. The licensee performed evaluation of these affected components for the power uprate condition. The calculated maximum ranges of stress intensities are provided in Tables 5.7-2, 5.7-3 and 5.7-4. The fatigue calculation were revised to reflect the power uprate condition. As a result of its

evaluation, the licensee indicated that the stress intensity ranges and fatigue usage factors provided are in compliance with the requirements of the ASME Code, Section III, 1986 Edition through the Winter 1987 Addenda, which is the Code of record at KNPP, and are, therefore, acceptable. The NRC staff concurs with the licensee's conclusion

In addition, the licensee evaluated the vibration of the U-bend tubes for Model 54F SGs at KNPP. The licensee indicated that the calculated fluid-elastic stability ratio is less than the allowable limit of 1.0, and that the maximum fluid induced displacement values due to turbulence and the vortex shedding are insignificant. As a result, the licensee concluded that the vibration of SG tubes will remain within the allowable limits for the power uprate. The NRC staff concurs with the licensee's conclusion.

On the basis of its review, the NRC staff concludes that the licensee has demonstrated the maximum stresses and CUFs for the limiting SG components to be within the Code allowable limits and, therefore, acceptable for the proposed 6.0-percent stretch power uprate.

3.4.2.5 Reactor Coolant Pumps

The licensee reviewed the existing design basis-analyses of the KNPP RCPs to determine the impact of the revised design conditions in Table 2.1-1. The licensee indicated that the KNPP RCPs predate the inclusion of pumps in the ASME Code, Section III, and are not Code stamped. Code editions used for the power uprate evaluation for the KNPP pumps range from the 1968 Edition with Winter 1970 Addenda, to the 1971 Edition with 1972 addenda.

After the power uprate, the RCS pressure remains unchanged. The licensee indicated that the design parameter of the RCP temperature (RPV inlet) as provided in Table 5.6-1 of the May 22, 2003, application for the power uprate condition is less than the present design basis. Also, there are no significant changes to the design thermal transients. CUFs for RCP limiting components shown in Table 5.6-3 are below the allowable limit of 1.0. Stresses for the RCP vertical and lateral supports as shown in Table 5.5-3 of the amendment request are less than the allowable. As a result of the evaluation, the licensee concluded that the current KNPP Model 93A RCPs remain in compliance with the applicable ASME Code requirements for structural integrity at the proposed power uprate.

On the basis of its review, the NRC staff concurs with the licensee's conclusion that the RCPs, when operating at the proposed uprated conditions with 6.0-percent power increase from the current rated power, will remain in compliance with the requirements of the codes and standards under which the KNPP were originally licensed.

3.4.2.6 Pressurizer

The licensee evaluated the limiting design locations of the pressurizer components. The components in the lower end of pressurizer (such as the surge nozzle, lower head well and penetration, and support skirt) are affected by the pressure and the hot-leg temperature. The components in the upper end of the pressurizer (such as the spray nozzle, instrument nozzle, safety and relief nozzle, and upper head and shell) are affected by the pressure and the cold-leg temperature for operation at the uprated conditions. The evaluation was performed using the ASME Code, Section III, 1965 Edition, through Summer 1966 addenda, which is the Code of record for KNPP pressurizer.

The key parameters in the current KNPP pressurizer stress report were compared against the revised design conditions in Table 2.1-1 for the proposed power uprate. The limiting operating conditions of the pressurizer occur when the RCS pressure is high and the RCS hot-leg (T_{hot}) and cold-leg (T_{cold}) temperatures are low. Because the proposed power uprate does not change the maximum RCS pressure and the pressurizer temperature (T_{sat}), the existing design-basis analyses with the lowest T_{hot}, that maximize thermal stresses in components at the lower end of the pressurizer, remain bounding for the proposed power uprate. However, there is a slight increase in thermal stress due to lower T_{cold} at the power uprate condition. The evaluation was performed to demonstrate the adequacy of the components in the upper end of the pressurizer. The calculated CUFs for limiting pressurizer locations at the uprated condition were found to be below the code allowable limit of unity as shown in Table 5.8-2, Appendix 4 of the May 22, 2003, application. As a result of the above evaluation, the licensee concluded that the existing pressurizer components will remain adequate for plant operation at the proposed 6.0-percent stretch power uprate while the RCS pressure remains unchanged. The NRC staff agrees with the licensee's conclusion and finds it acceptable.

3.4.2.7 Nuclear Steam Supplying System Piping and Pipe Supports

The proposed power uprate of KNPP involves the increase of temperature difference across the RCS. The licensee evaluated the NSSS piping and supports by reviewing the design-basis analysis against the uprated power design system parameters, transients and the LOCA dynamic loads. The evaluation was performed for the reactor coolant loop (RCL) piping, primary equipment nozzles, primary equipment supports, and the pressurizer surge line piping. USAS B31.1 Power Piping Code, 1967 Edition was used for the power uprate evaluation of KNPP RCS piping except the surge line which was evaluated in accordance with requirements of the ASME B&PV Code Section III, 1986 Edition, which is the Code of record. The Calculated stresses and CUFs are provided in Table 5.5.1-2 of the amendment request for the primary loop piping for the power uprate. The maximum calculated stresses and CUFs are shown less than the code allowable limits. In its response to the NRC staff's request for additional information, the licensee provided a summary of the stresses and CUF for the surge line and they are below the code allowable stress limits and the fatigue usage factor limit of 1.0.

The licensee also indicated that the design transients used in the evaluation of the RCS piping systems and equipment nozzles are unchanged for the KNPP power uprate. The proposed power uprate does not change the maximum RCS pressure. The design-basis LOCA forces due to postulated primary loop guillotine breaks have been eliminated using the loop leak-before-break (LBB) methodology for KNPP. With the use of LBB technology, LOCA forces for the power uprate condition were derived based on postulation of breaks in three branch lines at the surge line nozzle on the hot leg, the accumulator line nozzle at the cold-leg, and the RHR line nozzle on the hot-leg. As such, the design-basis LOCA hydraulic forcing functions are bounding for the LOCA loads at the uprated power condition. Furthermore, the deadweight and seismic loads are not affected by the power uprate. The licensee concluded that the existing stresses, fatigue usage factors and loads remain bounding for the power uprate for the NSSS components including the reactor cooling loop piping, the primary equipment nozzles, the primary equipment supports, pipe supports and the auxiliary equipment (i.e., heat exchangers, pumps, valves and tanks). Therefore, these components will continue to be in compliance with the Code of record at KNPP.

On the basis of its review of the licensee's submittals, the NRC staff concurs with the licensee's conclusion that the existing NSSS piping and supports, the primary equipment nozzles, the primary equipment supports, and the auxiliary lines connecting to the primary loop piping will remain in compliance with the requirements of the design bases criteria, as defined in the KNPP FSAR, and are therefore, acceptable for the proposed 6.0-percent stretch power uprate.

3.4.2.8 BOP Systems and Motor-Operated-Valves

The licensee evaluated the adequacy of the BOP systems based on comparing the existing design-basis parameters with the power uprate conditions. The BOP piping systems that were evaluated for the power uprate include main steam, condensate, feedwater, SG blowdown, and heater drains systems. The licensee evaluated these affected systems at the uprated power level by comparing the input parameters for the current piping analysis reports against the design parameters in Table 3.1-1 (e.g., RCS temperatures, and steam temperature and steam flow rate) for up to 1772 MWt reactor core power. In its response dated November 5, 2003, to the NRC staff's request for additional information, the licensee provided maximum calculated stresses for the above evaluated BOP piping to be less than the allowable limits. In its response, the licensee also indicated that piping and pipe support evaluations were not completed for the service water and CCW systems because there are open items to be resolved as mentioned in Appendix 7 to the May 22, 2003, amendment request. These open items are associated with the outlet temperatures for these two systems that are much higher than the design-basis temperatures in the existing analysis of record. As a result, the licensee concluded that with the exceptions of service water and CCW systems, the existing design-basis analyses for the BOP piping, pipe supports, and components will satisfy design-basis requirements when considering the temperature, pressure, and flow rate effects resulting from the proposed power uprate at KNPP. The licensee has made Commitment No. 7 in Enclosure 4 of its January 30, 2004, submittals: "Piping and pipe support evaluations concluded in Section 8.4.4 that systems remain acceptable assuming resolution of open items. Open items remained on the following systems: service water and CCW. These open items will be resolved prior to stretch uprate implementation."

The licensee also reviewed the programs, components, structures, and non-NSSS system issues as they relate to the power uprate. The NRC staff reviewed the licensee's evaluation, including specific examples of the effect of the power uprate on the functionality of safety-related pumps and valves at KNPP. The NRC staff also considered the results of its previous review of licensee programs at KNPP. For example, the inservice testing (IST) program for pumps and valves at KNPP will not be impacted because the power uprate will not modify the design function or system performance requirements of any safety-related systems or components. The safety-related pumps at KNPP have been evaluated to be adequately designed for operation at the requested power uprate conditions because the increased system flow remains within the original pump design capability. The only valve or piping configuration change necessary at KNPP to accommodate the power uprate was an adjustment to increase the flow capability of the main feedwater regulating valves. No changes to relief valve settings are required because the post uprate system operating pressures remain bounded by the pre-uprate system operating pressures.

As documented in a letter dated January 4, 1996, to M. L. Marchi from John M. Jacobson, the NRC completed a detailed review of the MOV program implemented at KNPP in response to GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," through inspection

activities and review of follow-up information. In a safety evaluation dated December 16, 1999, the NRC accepted the licensee's long-term MOV program in response to GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves," through review of information submitted by the licensee. In support of its power uprate request, the licensee evaluated the safety-related motor-operated valves (MOVs) at KNPP and found that the assumed fluid system parameters bound the power uprate parameters. With respect to the power uprate post-accident environmental conditions, the assumed post-accident conditions for the safety-related MOVs inside containment at KNPP bound the power uprate conditions. For safety-related MOVs outside containment, the licensee committed to revise the degraded voltage and thrust calculations, as necessary, to reflect the power uprate post-accident temperatures. The licensee has made Commitment No. 16 in Enclosure 4 of its January 30, 2004, submittals: "The response to RAI question #57 to the stretch power uprate submittals stated degraded voltage and thrust calculations for MOV operators outside containment were reviewed for impact of the uprated post accident temperatures, and will be revised, as required, prior to implementation of the 6 percent stretch power uprate."

The licensee's programs at KNPP for air-operated valves and check valves were not impacted by the power uprate because the system design capacities assumed in those programs bound the power uprate parameters. The licensee also evaluated the impact of the power uprate on its response to GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves." The power-operated gate valves evaluated in response to GL 95-07 will not be adversely affected by potential pressure locking or thermal binding associated with the power uprate, because the power uprate conditions are bounded by existing program assumptions. The NRC staff finds the licensee's evaluation of the effect of the proposed power uprate on the capability of safety-related pumps and valves at KNPP to be acceptable, based on the NRC staff's review of the licensee's programs in response to GLs 89-10, 95-07, and 96-05; and the review of the information submitted by the licensee describing the scope, extent, and results (with specific examples) of the evaluation of safety-related pumps and valves at KNPP.

The licensee reviewed the evaluation of NMC GL 96-06 program regarding the over-pressurization of isolated piping segments. The licensee concluded that the existing evaluation for GL 96-06 was performed. The licensee found that the existing analysis is bounding for the proposed power uprate condition. On the basis of the above review, the NRC staff concurs with the licensee's conclusions that the power uprate will have no adverse effects on the safety-related valves and that conclusions of the NMC GL 95-07, and GL 96-06, as well as GL 89-10 programs, remain valid.

As a result of the above evaluation, the NRC staff concludes that the BOP piping, pipe supports and equipment nozzles, and valves remain acceptable and continue to satisfy the design-basis requirements for the proposed 6.9-percent stretch power uprate.

3.4.3 Conclusion

The NRC staff has reviewed the licensee's evaluation of the impact of the proposed stretch power uprate on NSSS and BOP systems and components with regard to stresses, CUFs, vibration, high-energy line break (HELB) locations, jet impingement and thrust forces, and safety-related valve programs and concludes that the these areas will continue to be acceptable following implementation of the proposed stretch power uprate. Therefore, the NRC

staff finds the proposed 6.0-percent stretch power uprate acceptable with respect to the areas of civil and mechanical engineering.

3.5 Dose Consequences Analysis

3.5.1 Regulatory Evaluation

The NRC staff review covers the impact of the proposed stretch power uprate on the results of dose consequence analyses. The regulatory requirements for which the NRC staff based its acceptance are the accident dose criteria in 10 CFR 50.67, as supplemented in Regulatory Position 4.4 of RG 1.183, “Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors,” and 10 CFR Part 50 Appendix A, GDC-19, “Control Room,” as supplemented by Section 6.4 of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (SRP). Except where the licensee proposed a suitable alternative, the NRC staff utilized the regulatory guidance provided in SRP Section 15.0.1, “Radiological Consequence Analysis Using Alternative Source Terms,” in performing the review. The NRC staff also considered relevant information in the KNPP USAR, TSs, and recently approved amendments. Please note that Tables 1 through 10 referenced in the subsections below are provided in Attachment 2 to this SE.

3.5.2 Technical Evaluation

3.5.2.1 Technical Evaluation Scope

The NRC staff reviewed the regulatory and technical analyses, as related to the radiological consequences of design-basis accidents, performed by NMC in support of its proposed license amendment. Information regarding these analyses was provided in Section 2 and Attachment 4 (proprietary) and 5 (non-proprietary) of the May 22, 2003, application. The NRC staff also reviewed the licensee’s letter dated January 30, 2004, which included a response to a request for additional information by the NRC staff. The NRC staff reviewed the assumptions, inputs, and methods used by NMC to assess the impact of the requested power uprate on the radiological consequences of design-basis accidents. The findings of this SE are based on the descriptions of the licensee’s analyses and other supporting information docketed by NMC. Only docketed information was relied upon in making this safety finding.

The licensee re-analyzed the radiological consequences for the following eight design-basis accidents (DBAs) to account for the uprated power:

- Main steamline break (MSLB)
- Locked reactor coolant pump rotor
- Rod ejection
- SG tube rupture (SGTR)
- Large-break loss-of-coolant accident (LBLOCA)
- Fuel-handling accident (FHA)
- Waste gas decay tank (GDT) rupture
- Volume control tank (VCT) rupture

These DBAs were previously analyzed in the licensing submittals to support Amendment No. 166, issued March 17, 2003 (ADAMS Accession No. ML030210062), to the KNPP license,

which implemented an AST in accordance with 10 CFR 50.67. These previously approved radiological analyses used the analytical methods and assumptions outlined in RG 1.183. The revised analyses submitted to support the current stretch power uprate to 1772 MWt are substantially the same as the previously approved analyses, with changes as discussed below.

The revised core and coolant inventories are based on power operation at 1772 MWt increased by 0.6 percent to 1782.6 MWt to account for power measurement uncertainty.

3.5.2.2 Main Steamline Break Accident

The licensee assumed that the faulted SG boils dry within 2 minutes. The entire liquid inventory of the faulted SG is steamed off and all the iodine initially in the SG is released to the outside environment. The primary-to-secondary steam generator tube leakage rate is assumed to be at the TS limit of 150 gallons per day (gpd) per SG. The 150 gpd leakage for the faulted SG, along with its noble gas and iodine, is assumed released directly to the outside atmosphere. In the intact SG, the 150 gpd primary-to-secondary leakage mixes with the bulk SG secondary coolant water. Transferred noble gases are released without holdup, and iodine is released to the outside environment at the steaming rate of the intact SG, with credit for partitioning when the SG tubes are covered with water.

NMC analyzed the MSLB for two iodine spiking cases. The pre-accident iodine spike case assumed that a reactor transient has occurred prior to the MSLB, and has raised the RCS iodine concentration to the TS limit for a transient of 60 microcuries per gram (µCi/gm) of dose equivalent (DE) Iodine 131 (I-131). The accident-initiated iodine spike case assumed that the reactor trip associated with the MSLB creates an increase in the iodine release rate from the fuel to the coolant to a value 500 times greater that the release rate corresponding to a maximum equilibrium RCS concentration of 1.0 µCi/gm DE I-131. The accident-initiated spike duration is 4 hours. The secondary coolant activity in both cases is assumed to be the TS limit of 0.1 µCi/gm DE I-131. No fuel damage is projected for the MSLB.

The low steamline pressure safety injection (SI) setpoint will be reached shortly after the onset of an MSLB. The SI signal causes the control room heating, ventilation and air conditioning (HVAC) to switch from normal-operation mode to the accident mode of operation. NMC conservatively assumed that the control room HVAC does not fully enter the accident mode of operation until 5 minutes after the event begins. The NRC staff finds this assumption to be acceptable based on the operation of the reactor protection system and control room HVAC system.

The NRC staff reviewed the licensee’s methods and assumptions used in its analysis of the MSLB radiological consequences performed for the 6.0-percent stretch uprate, and finds that they are consistent with the conservative guidance provided in RG 1.183. The licensee’s calculated radiological consequences at the Exclusion Area Boundary (EAB), Low Population Zone (LPZ) and in the KNPP control room are within the dose criteria specified in 10 CFR 50.67 and GDC-19 and are within the acceptance criteria given in SRP 15.0.1 for the MSLB. The results of the licensee’s calculations are provided in Table 1, and the major parameters and assumptions used by the licensee and acceptable to the NRC staff are listed in Table 2.

3.5.2.3 Locked RCP Rotor Accident

Although the licensee's USAR analysis predicts that less than 50 percent of the fuel rods in the core undergo DNB and may be damaged, the licensee conservatively assumed that all of the fuel rods in the core are damaged with all of the fuel gap activity released to the RCS. This is more conservative than the guidance in RG 1.183, and the NRC staff finds it acceptable. The licensee's analysis also assumes that a reactor transient has occurred prior to the locked rotor event that has raised the RCS iodine activity concentration to 60 µCi/gm DE I-131. The noble gas and alkali metal activity concentration in the primary coolant is based on a fuel defect level of 1 percent. The iodine activity concentration in the secondary coolant is assumed to be 0.1 µCi/gm DE I-131, and the alkali metal activity concentration is assumed to be 10 percent of the primary coolant concentration. Accident induced activity is assumed to be released to the environment as a result of primary-to-secondary leakage through the SG tubes and steaming from the secondary side, released through either the atmospheric relief valves or safety valves. An iodine partitioning factor in the SGs of 0.01 is used to account for retention of iodine in the SG as the water turns to steam. The partitioning factor of 0.01 is also applied to the alkali metal activity release. All noble gas activity carried over to the secondary side of the SGs is assumed to be immediately released to the outside atmosphere. At 8 hours after the accident, the licensee assumed that RHR system has removed all decay heat with no further releases to the environment after that time. The NRC staff finds these assumptions are consistent with guidance in RG 1.183 and are acceptable.

NMC assumed that the control room HVAC system is in normal-operation mode at the onset of the accident. A high-radiation signal for the air supply duct is generated as a result of the activity release to the atmosphere. Control room HVAC is not assumed to fully enter the accident mode of operation until 10 minutes after the event begins. The staff finds these assumptions to be acceptable based on the operation of the radiation monitoring and control room HVAC systems.

The NRC staff reviewed the licensee's methods, inputs and assumptions used in its radiological consequences analysis of the locked-rotor accident for the 6.0-percent stretch power uprate, and finds that they are consistent with or more conservative than the guidance given in RG 1.183. The licensee's calculated radiological consequences at the EAB, LPZ and in the KNPP control room are within the dose criteria specified in 10 CFR 50.67 and GDC-19, and are within the acceptance criteria given in SRP 15.0.1 for the locked rotor accident. The results of the licensee's calculations are provided in Table 1, and the major parameters and assumptions used by the licensee and acceptable to the NRC staff are listed in Table 3.

3.5.2.4 Control Rod Ejection Accident

This DBA postulated the mechanical failure of a CRDM pressure housing that results in the ejection of a RCCA and drive shaft. Localized damage to fuel cladding and a limited amount of fuel melting are projected. The radioactivity in the primary coolant is assumed to leak through the SG tubes into the secondary coolant. A portion of this activity is released to the outside atmosphere through the main condenser, atmospheric relief valves or safety valves. Additionally, radioactive primary coolant is discharged to the containment through the opening in the reactor vessel head where the control rod assembly was ejected. The activity in the

containment is assumed released to the outside atmosphere as a result of design-basis containment leakage of 0.5 percent per day for the first 24 hours. After that, the containment is assumed to leak at half that rate until the end of the 30-day period considered in the analysis.

NMC assumed that 15 percent of the fuel rods in the core suffer sufficient damage such that all their gap activity is released. The licensee assumed that 10 percent of the total core activity of iodine and noble gases and 12 percent of the total core activity for alkali metals are in the fuel gap, consistent with guidance provided in RG 1.183. A small fraction of the fuel in the failed rods is assumed to melt as a result of the rod ejection. The licensee estimated this melting to be limited to 0.375 percent of the core. This estimate was previously found acceptable by the NRC staff in the SE for Amendment No. 166. The requested 6.0-percent stretch power uprate would not affect any of the parameters that were used to develop the fuel melting estimate. Therefore, the NRC staff finds that the rod ejection fuel damage assumptions remain acceptable.

The licensee assumed 100 percent of noble gases and alkali metals in the failed fuel gap and melted fuel are released to either the RCS or the containment, depending on the pathway assumed. For the containment leakage pathway, the licensee assumed that all the iodine from the gap of the failed fuel and 25 percent of the iodine released from melted fuel are released to the containment atmosphere. For the primary-to-secondary leakage release pathway, the licensee assumed that all the iodine from the gap of the failed fuel and 50 percent of the iodine released from melted fuel is released to the RCS. These assumptions are consistent with the guidance in RG 1.183, and are acceptable to the NRC staff.

As discussed above for the locked rotor accident, the licensee assumed an iodine partitioning factor of 0.01 in the SGs for the primary-to-secondary leakage release pathway. For the containment leakage release pathway, no credit was taken for iodine or particulate removal mechanisms. The NRC staff finds that this is consistent with guidance in RG 1.183 and is acceptable.

The low pressurizer pressure SI setpoint is expected to be reached within 60 seconds of the onset of the control rod ejection. The SI signal causes the control room HVAC to switch from the normal-operation mode to the accident mode of operation. The licensee conservatively assumed that the control room HVAC does not fully enter the accident mode of operation until 2.5 minutes after the event begins. The NRC staff finds these assumptions to be acceptable based on the operation of the reactor protection and control room HVAC systems.

The NRC staff reviewed the licensee’s methods, inputs and assumptions used in its radiological consequences analysis of the control rod ejection accident for the 6.0-percent stretch power uprate and finds that they are consistent with the conservative guidance given in RG 1.183. The licensee’s calculated radiological consequences at the EAB, LPZ and in the KNPP control room are within the dose criteria specified in 10 CFR 50.67 and GDC-19 and are within the acceptance criteria given in SRP 15.0.1 for the control rod ejection accident. The results of the licensee’s calculations are provided in Table 1, and the major parameters and assumptions used by the licensee and acceptable to the NRC staff are listed in Table 4.

3.5.2.5 Steam Generator Tube Rupture

The SGTR is analyzed for two iodine spiking cases; a pre-existing iodine spike results in elevated primary coolant activity, or an iodine spike is assumed to be initiated by the accident. For the pre-existing iodine spike case, the RCS iodine activity concentration is assumed to be at the TS limit for a transient of 60 µCi/gm DE I-131. For the accident initiated iodine spike case, the associated reactor trip causes an increase in the iodine release rate from the fuel to the RCS to a value 500 times the rate associated with the TS equilibrium RCS activity concentration of 1.0 µCi/gm DE I-131. The duration of the accident initiated iodine spike is limited by the amount of iodine in the fuel gap. Based on having 8 percent of the core inventory of iodine in the fuel gap, the spike would last 4 hours. RG 1.183 allows an accident initiated spiking factor of 335 for the SGTR, and the NRC staff finds the licensee's assumed factor of 500 is conservative compared to the RG value. All other analysis inputs are consistent with the guidance in RG 1.183.

The low pressurizer pressure SI setpoint is expected to be reached at around 2.9 minutes after the onset of the SGTR. The SI signal causes the control room HVAC to switch from the normal-operation mode to the accident mode of operation. The licensee conservatively assumed that the control room HVAC does not fully enter the accident mode of operation until 5 minutes after the event begins. The NRC staff finds these assumptions to be acceptable based on the operation of the reactor protection and control room HVAC systems.

The NRC staff reviewed the licensee's methods, inputs and assumptions used in its radiological consequences analysis of the SGTR for the 6.0-percent stretch power uprate and finds that they are consistent with the conservative guidance given in RG 1.183. The licensee's calculated radiological consequences at the EAB, LPZ and in the KNPP control room are within the dose criteria specified in 10 CFR 50.67 and GDC-19 and are within the acceptance criteria given in SRP 15.0.1 for the SGTR. The results of the licensee's calculations are provided in Table 1, and the major parameters and assumptions used by the licensee and acceptable to the NRC staff are listed in Table 5.

3.5.2.6 Large-Break Loss-of-Coolant Accident

The licensee's analysis of the LBLOCA radiological consequences uses analytical methods and assumptions consistent with the guidance in RG 1.183. The analysis for the 6.0-percent stretch power uprate follows the methods found acceptable by the NRC staff in the SE for Amendment No. 166 issued March 17, 2003, to the KNPP license, which implemented an AST in accordance with 10 CFR 50.67, with changes made to account for operation at the increased power. All source term assumptions used by the licensee are consistent with guidance in RG 1.183 and will not be reiterated here. Activity from the damaged core is released into the containment. Three pathways for release to the environment are considered in the analysis:

(1) design-basis containment leakage,
(2) leakage from ESF systems outside containment, and
(3) ECCS recirculation back-leakage to the RWST.

The calculated radiological consequences of these three release pathways are added together to determine the total LBLOCA radiological consequences.

3.5.2.6.1 Containment Leakage Pathway

The containment is assumed to leak at the design-basis leak rate of 0.5 percent per day for the first 24 hours of the accident, and then to leak at half that rate for the remainder of the 30-day analysis period. The licensee assumed that during the first 10 minutes of the accident, 90 percent of the activity leaking from the containment is discharged directly to the environment. The remaining 10 percent enters the auxiliary building where it is released through filters. After 10 minutes, only 1 percent of the activity leaking from the containment is assumed to go directly to the environment, 10 percent continues to go to the auxiliary building, and 89 percent is assumed to go into the shield building. The air discharged from the shield building is filtered. Additionally, once the shield building is brought to sub-atmospheric pressure at 30 minutes into the accident, iodine and particulates can be removed by recirculation through filters. A shield building participation fraction of 0.5 is assumed. The NRC staff found these assumptions acceptable in the SE for Amendment No. 166.

The licensee assumed removal of iodine through sedimentation for particulates and the containment spray for elemental and particulate forms of iodine. The KNPP containment spray system is an ESF system and is designed to provide containment cooling and fission product removal in the containment following a LOCA. One train of spray was assumed to operate. Switchover to recirculation spray is not credited and all spray removal is terminated when the RWST drains down at 0.91 hours from the start of the accident. In determining the core spray iodine removal rates, the licensee assumed a reduction in assumed spray flow relative to that assumed in the analysis supporting Amendment No. 166. This reduction is intended to bound potential pump degradation. The NRC staff finds this change acceptable. The licensee assumed a sedimentation coefficient of 0.1 hr^{-1} for particulates after the core spray system is terminated. This assumption was found acceptable to the NRC staff in the SE for Amendment No. 166 to the KNPP license. The licensee used the models and guidance provided in RG 1.183 and SRP 6.5.2, “Containment Spray as a Fission Product Cleanup System,” to determine the removal rates for iodine. Therefore, the NRC staff finds the removal rates calculated by the licensee are acceptable.

3.5.2.6.2 Post-LOCA ESF Leakage Pathway

During the recirculation phase of long-term core cooling, radioactive water from the containment sump is sent to ECCS equipment located outside the containment. These components may leak into the auxiliary building. Although ECCS recirculation does not occur until 0.91 hours after the accident begins, the licensee conservatively assumed leakage occurs immediately upon the onset of the LBLOCA. The licensee conservatively assumed the leakage to the auxiliary building is 12 gallons per hour. This is twice as much as assumed currently at KNPP. The licensee assumed that 10 percent of the activity in the leaked fluid becomes airborne when the sump temperature is above 212 °F. Once the sump temperature drops below 212 °F at 3 hours from the start of the event, the airborne activity fraction is reduced to 1 percent of the activity in the leaked fluid. The assumed time that the sump temperature falls below 212 °F is selected to bound the results of the containment response analyses performed for the 6.0-percent stretch power uprate. The NRC staff finds these assumptions are consistent with RG 1.183 and are acceptable. The licensee also assumed that half of the airborne iodine activity in the auxiliary building is removed by plateout on surfaces. This assumption was found to be acceptable to the NRC staff in the SE for Amendment No. 166 to the KNPP license.

3.5.2.6.3 ECCS Back-Leakage to the RWST

RHR back-leakage to the RWST is assumed at a rate of 3 gpm for the first 24 hours, and 1.5 gpm for the remainder of the accident. It is assumed that 1 percent of the iodine in the leakage becomes airborne, even when the sump temperature is above 212 °F since any incoming water would be cooled by the water remaining in the RWST. This assumption was found to be acceptable to the NRC staff in the SE for Amendment No. 166 to the KNPP license.

3.5.2.6.4 Control Room Ventilation System Modeling

For the LBLOCA, the low pressurizer pressure SI setpoint will be reached shortly after the start of the event. The SI signal causes the control room HVAC to switch from the normal-operation mode to the accident mode of operation. The licensee conservatively assumed that the control room HVAC does not fully enter the accident mode of operation until 2 minutes after the event begins. The NRC staff finds these assumptions to be acceptable based on the operation of the reactor protection and control room HVAC systems.

3.5.2.6.5 LBLOCA Conclusion

The NRC staff reviewed the licensee's methods, inputs and assumptions used in its radiological consequences analysis of the LBLOCA for the 6.0-percent stretch power uprate and finds that they are consistent with the conservative guidance given in RG 1.183. To verify the licensee's dose results, the NRC staff performed confirmatory radiological consequences analyses of the LOCA. The licensee's calculated radiological consequences at the EAB, LPZ and in the KNPP control room are within the dose criteria specified in 10 CFR 50.67 and GDC-19 and are within the acceptance criteria given in SRP 15.0.1 for the LOCA. The results of the licensee's calculations are provided in Table 1, and the major parameters and assumptions used by the licensee and acceptable to the NRC staff are listed in Table 6.

3.5.2.7 Fuel Handling Accident

The licensee's analysis of the FHA was performed with assumptions selected so that the results are bounding for the accident that occurs either in the containment or in the auxiliary building. Activity released from the damaged assembly is assumed to be released to the environment through either the containment purge system or the spent fuel pool ventilation system, without credit for filtration or isolation or the containment, containment purge system, or spent fuel pool ventilation system. The decay time used is the TS minimum decay time before movement of fuel of 100 hours. The licensee assumed that all the fuel rods in the equivalent of one fuel assembly are damaged, and all the gap activity in the rods is released to the pool. A pool iodine effective decontamination factor of 200 is assumed. All fuel gap noble gas activity is assumed released from the pool. All activity released from the pool is assumed to be released to the outside environment within 2 hours.

NMC assumed that the control room HVAC system is in normal-operation mode at the onset of the FHA. A high-radiation signal for the air supply duct is generated almost immediately as a result of the activity release to the atmosphere. Control room HVAC is not assumed to fully enter the accident mode of operation until 1 minute after the event begins. The NRC staff finds these assumptions to be acceptable based on the operation of the radiation monitoring and control room HVAC systems.

The NRC staff reviewed the licensee's methods, inputs and assumptions used in its radiological consequences analysis of the FHA for the 6.0-percent stretch power uprate and finds that they are consistent with the conservative guidance given in RG 1.183. To verify the licensee's dose results, the NRC staff performed confirmatory radiological consequences analyses of the FHA. The licensee's calculated radiological consequences at the EAB, LPZ and in the KNPP control room are within the dose criteria specified in 10 CFR 50.67 and GDC-19 and are within the acceptance criteria given in SRP 15.0.1 for the FHA. The results of the licensee's calculations are provided in Table 1, and the major parameters and assumptions used by the licensee and acceptable to the NRC staff are listed in Table 7.

3.5.2.8 Waste Gas Decay Tank Rupture/Volume Control Tank Rupture

The KNPP licensing basis includes analyses of the radiological consequences of rupture of the waste GDT and rupture of the VCT. SRP 15.0.1 does not include guidance on review of these accidents. The offsite dose acceptance criterion for the GDT rupture was previously given as 500 mrem whole body in Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," (attached to SRP 11.3, Rev. 0, July 1981). Although the licensee's May 22, 2003, application states a licensee dose acceptance criterion of 6.3 rem TEDE for these accidents, the NRC staff does not find that criterion acceptable. In a January 30, 2004, response to an NRC request for additional information, the licensee stated that they did not intend to request a higher dose acceptance criterion be accepted by the NRC staff and that the original licensing basis applies. For the 6.0-percent stretch power uprate, the licensee's analyses of these accidents show that they continue to meet the current KNPP licensing basis.

For the GDT rupture, the licensee assumed that the inventory of gases in the GDT are based on plant operation with 1 percent failed fuel, and with no purge of activity from the VCT to the GDT during the plant operating cycle. The GDT rupture release duration is 5 minutes. The licensee assumed that the control room HVAC system begins in normal-operation mode. A high-radiation signal for the air supply duct is generated almost immediately as a result of the activity release to the atmosphere. Control room HVAC is not assumed to fully enter the accident mode of operation until 30 seconds after the event begins. Because of the short duration of the radiation release, minimizing the assumed control room unfiltered inleakage maximizes the calculated control room dose. This is due to less dilution of the radioactivity in the control room. The NRC staff finds these assumptions to be acceptable based on plant operation and the operation of the radiation monitoring and control room HVAC systems.

For the VCT rupture, the licensee assumed that the inventory of gases in the VCT are based on plant operation with 1 percent failed fuel, 90 percent of the iodine removed by the letdown demineralizer, and with no purge of the VCT gas space. During the event, letdown flow to the VCT continues at the maximum flow rate of 88 gpm for 5 minutes, when the letdown line is assumed to be isolated. The primary coolant iodine activity is assumed to be at the pre-existing iodine spike level of 60 µCi/gm DE I-131, which is reduced by 90 percent by the letdown demineralizer. All of the noble gas and 1 percent of the iodine in the letdown flow is assumed to be released to the environment. The licensee assumed that the control room HVAC system begins in normal-operation mode. A high-radiation signal for the air supply duct is generated almost immediately as a result of the activity release to the atmosphere. Control room HVAC is not assumed to fully enter the accident mode of operation until 30 seconds after the event begins. Because of the short duration of the radiation release, minimizing the assumed control

room unfiltered inleakage maximizes the calculated control room dose. This is due to less dilution of the radioactivity in the control room. The NRC staff finds these assumptions to be acceptable based on plant operation and the operation of the radiation monitoring and control room HVAC systems.

The NRC staff reviewed the licensee's methods, inputs and assumptions used in its radiological consequences analysis of the GDT and VCT rupture accidents for the 6.0-percent stretch power uprate. The assumptions were previously found acceptable by the NRC staff in its SE for Amendment No. 166 to the KNPP license. The only assumptions affected by the 6.0-percent stretch power uprate are the tank and coolant activity inventories. The NRC staff finds the analyses acceptable. The radiological consequences calculated by the licensee for the GDT rupture and the VCT rupture are small fractions (less than 1 percent) of the dose criteria specified in 10 CFR 50.67 and are well below the dose criteria in GDC-19 for control room habitability, and therefore are acceptable to the NRC staff. The licensee's analyses show that the original licensing basis for the GDT and VCT rupture continues to be met for the 6.0-percent stretch power uprate. The results of the licensee's calculations, and the major parameters and assumptions used by the licensee and acceptable to the NRC staff are listed in Tables 8 and 9.

3.5.2.9 Control Room Habitability

In their analyses, NMC assumed that the control room unfiltered inleakage was 200 cfm. This analysis assumption was based on air flow measurements and was accepted by the NRC staff for use as documented in the SE for Amendment No. 166 to the KNPP license. NMC has not performed integrated leakage testing to confirm this leakage value. On June 12, 2003, the NRC staff issued GL2003-01, "Control Room Habitability." This GL identifies the NRC staff concerns regarding the reliability of current surveillance testing to identify and quantify control room inleakage, and requests licensees to confirm the most limiting unfiltered inleakage into their control room envelope. NMC is required by the GL to respond to the information request within 180 days of its issue. By letter dated August 7, 2003, the licensee submitted a 60-day response to GL 2003-01, in which they committed to provide a schedule for providing the information requested, including performance of American Society for Testing and Materials (ASTM) E741 (tracer gas) testing of the KNPP control room envelope. By letter dated November 25, 2003, the licensee committed to perform testing of the KNPP control room envelope to determine unfiltered inleakage in December 2004. The licensee also committed to providing, within 90 days after testing, confirmation that the most limiting unfiltered inleakage into the KNPP control room envelope is no more than the value assumed in its design-basis radiological analyses for control room habitability, as requested in GL 2003-01 item 1(a).

The NRC staff has determined that there is reasonable assurance that the KNPP control room will be habitable during a DBA and this amendment may be approved prior to the NRC staff's review of the NMC response to the GL. The NRC staff bases this determination on (1) the relative magnitude of the infiltration assumed in the KNPP analyses, (2) the results of the periodic surveillance testing that has been performed and, (3) the low probability of occurrence in the interim period of the postulated accidents that could result in radioactivity releases sufficient to challenge the ability of the control room operators to protect the health and safety of the public. The NRC staff's approval of this amendment does not relieve NMC of addressing

the information requests in GL 2003-01 and does not imply that the NRC staff would necessarily find the analysis in this amendment acceptable as a response to information request 1(a) in GL 2003-01.

3.5.2.10 Technical Support Center

NUREG-0737, "Clarification of TMI Action Plan Requirements," Section III.A.1.2, requires that the technical support center (TSC) shall be habitable to the same degree as the control room for postulated accident conditions. The licensee stated that the original DBA evaluations of the control room and TSC indicated that the control room was bounding. However, the control room model has been revised for the 6.0-percent stretch power uprate and AST. NMC performed a comparison of the control room model to the TSC model, assuming the updated assumptions on unfiltered inleakage and recirculation flow for the control room, while the rest of the assumptions and inputs remained as in the original evaluation. The licensee's comparison evaluation confirmed that the TSC doses for the requested 6.0-percent stretch power uprate would be less than those calculated for the control room. The NRC staff finds that the licensee's evaluation of the TSC is reasonable and there is reasonable assurance that the TSC meets its requirements.

3.5.3 Conclusion

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by NMC to assess the radiological impacts of a 6.0-percent stretch power uprate at the KNPP. The NRC staff finds that NMC used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 3.5.1 above. The NRC staff compared the doses estimated by NMC to the applicable criteria identified in Section 3.5.1. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and control room doses will continue to comply with these criteria. Therefore, uprating the licensed core power level to 1772 MWT is acceptable with regard to the radiological consequences of postulated design-basis accidents.

3.6 <u>Materials and Chemical Engineering</u>

3.6.1 Regulatory Evaluation

The NRC staff review in the area of materials and chemical engineering covers reactor vessel integrity, SG tube integrity, and erosion corrosion programs. The NRC staff's review in this area focuses on the impact of proposed stretch power uprate on pressurized thermal shock calculations, fluence evaluations, heatup and cooldown P-T limit curves, low-temperature overpressure protection, upper-shelf energy, surveillance capsule withdrawal schedules, licensee programs for addressing SG tube degradation mechanisms, and erosion/corrosion. This review is conducted to verify that the results of licensee analyses related to these areas continue to meet the requirements of 10 CFR 50.60, 10 CFR 50.61, and 10 CFR 50.55a; and 10 CFR Part 50, Appendices G and H, following implementation of the proposed stretch power uprate. Additional guidance for the NRC staff's review of the topics within the materials and chemical engineering area include the guidance contained in Chapters 4, 5, and 6 of NUREG-0800.

3.6.2 Technical Evaluation

3.6.2.1 Reactor Pressure Vessel

Regarding the KNPP RPV surveillance program and capsule withdrawal schedule, the licensee concluded in Section 5.1.2.5:

> A calculation of [change of reference temperature nil ductility] ΔRT_{NDT} at 33 [effective full power years] EFPY was performed to determine if the increased fluences alter the number of capsules to be withdrawn from Kewaunee. This calculation determined that the maximum ΔRT_{NDT} using the uprated fluence for Kewaunee at [end-of-life] EOL is greater than 200^0 F. These ΔRT_{NDT} values would require five capsules to be withdrawn from Kewaunee. However, due to changes in capsule fluence, Capsule T should be removed before it receives a fluence of $7.12x10^{19}$ [neutrons per square centimeter] n/cm^2 (E>1.0 MeV [million electron volts]) (i.e., twice the peak vessel EOL fluence of $3.56x10^{19}$ n/cm^2 (E>1.0 MeV). This capsule may be held without testing following withdrawal.

Consistent with the requirements of Appendix H to 10 CFR Part 50, the licensee utilizes the guidance in the 1982 edition of ASTM E185 (E185-82) to define the number of surveillance capsules in the KNPP surveillance capsules in the KNPP surveillance program and their withdrawal requirements. For RPVs which demonstrate shifts in material transition temperature (ΔRT_{NDT}) in excess of 200 ^{0}F, E185-82 requires that the surveillance program have 5 capsules in it with the last pulled at a fluence between one and two times the end of license fluence at the RPV inside diameter. ASTM E185-82 permits the last surveillance capsule to be held without testing following withdrawal. Since the licensee's RPV surveillance capsule withdrawal schedule will continue to be consistent with the provisions of Appendix H to 10 CFR Part 50 and ASTM E185-82, the NRC staff finds the licensee's program to be acceptable.

Regarding the topic of the RPV PT limits, the licensee concluded in Section 5.1.2.5 that:

> This review indicates that the revised ART after the power uprate program will be more restrictive than that used in developing the current ART values for Kewaunee at 33 EFPY. Therefore, a change in applicability date is required. The 33 EFPY PT curves for Kewaunee will be applicable to 31.1 EFPY after the uprating.

The KNPP TSs contain 33 EFPY PT limit curves. Based on the uprated fluence, the current vessel EOL (33 EFPY) fluence will be reached at 31.1 EFPY. Hence, the ART value upon which the existing PT limit curves are based will also be reached at 31.1 EFPY instead of 33 EFPY, and the current PT limits will be applicable up to 31.1 EFPY. Therefore, the NRC staff concludes that the licensee's proposal to limit the existing heatup and cooldown curves to a period of applicability through 31.1 EFPY of operation is acceptable and consistent with the requirements of Appendix G to 10 CFR Part 50. The licensee has updated the technical specification accordingly.

Regarding the topics of PTS and upper shelf energy (USE) analyses for the KNPP RPV, the licensee provided the reference temperature PTS (RT_{PTS}) and USE values for the beltline materials of the KNPP vessel and concluded in its May 22, 2003, application. Regarding PTS, the calculated neutron fluence values for the Power Uprate Program condition at KNPP have increased over the current fluence. The RT_{PTS} calculation for the circumferential weld has used

master curve technology using NRC-approved methods. Based on licensee's evaluation all RT_{PTS} values will remain below the NRC screening criteria values using projected Power Uprate Program fluence through EOL (33 EFPY). Regarding USE, the revised fluence projections associated with the Power Uprate Program have increased the fluence projections used in developing the current predicted EOL USE values. All USE values for KNPP will maintain a level above 50 ft-lb. level at end of license (33 EFPY).

The NRC staff has evaluated the information provided by the licensee as well as information contained in the staff's Reactor Vessel Integrity Database. The NRC staff has also used NRC Master Curve Technology in evaluating the PTS. The NRC staff has previously approved the Master Curve technology for use to KNPP vessels (dated May 1, 2001, ADAMS Accession No. ML011210180). Based on the revised fluence values noted and the approved NRC method of Master Curve technology, the NRC staff independently confirmed that the Kewaunee RPV materials would continue to meeting the PTS screening criteria requirements of 10 CFR 50.61. The NRC staff has also verified USE requirements of Appendix G to 10 CFR Part 50 though EOL.

Regarding the integrity of reactor vessel internals, the licensee provided information regarding changes to operating temperature, flow rates, and neutron fluences which result from the proposed stretch power uprate. The licensee's evaluations of the critical components indicated that the structural integrity of the reactor internals will be maintained at the stretch uprated RCS conditions.

Based on the information provided by the licensee regarding insignificant changes to operating temperature, flow rates, and neutron fluences which result from the proposed stretch power uprate, the NRC staff agrees that the integrity of the RPV internals will be maintained such that the licensee's ability to meet the regulatory requirements in 10 CFR 50.46 regarding ECCS performance and maintaining a coolable core geometry will not be adversely impacted.

3.6.2.2 Flow-Accelerated Corrosion Program

Flow-accelerated corrosion (FAC) is a corrosion mechanism causing wall thinning of high energy pipes in the power conversion system which may lead to their failure. Since failure of these pipes may result in undesirable challenges to the plant's safety systems, the licensee has a program for predicting, inspecting, and repairing or replacing the components whose wall thinning exceeds the values required for their safe operation. In the submittals, the licensee stated predictive analysis was performed for the 6.0-percent stretch power uprate using the CHECWORKs computer code developed by the Electric Power Research Institute. Although wear rates increased in some lines, the identified changes were not significant and were not projected to cause wear rates or inspection intervals to change significantly. The licensee stated the KNPP FAC program will incorporate the increase in flow rate and velocities as well as changes in operating pressures and temperatures. The licensee committed to update the KNPP FAC models prior to the next programs inspections scheduled for the next refueling outage. The NRC staff considers this licensee's action adequate for ensuring integrity of the high energy pipes and therefore finds the licensee evaluation to be acceptable.

3.6.2.3 Structural Integrity and Primary-to-Secondary Pressure Differential Evaluation

The licensee performed a power uprate evaluation for the structural integrity of the SGs based on the existing analysis from a previous KNPP replacement steam generator project (RSG) that took place in 2001. The majority of the structural analysis performed in support of the KNPP RSG project remained applicable or bounding for the proposed 6.0-percent uprated condition. The components that required further evaluation were the feedwater nozzle, thermal sleeve, and J-nozzle-to-feedring weld. Stress and fatigue analysis results from the limiting locations in the feedwater nozzle, thermal sleeve, and J-nozzle-to-feedring weld were shown to remain within the ASME Code allowable limits.

Maximum primary-to-secondary side pressure differentials under normal conditions and upset transient conditions were analyzed and compared to the design pressure requirements in ASME B&PV Code, Section III. This analysis was performed using stretch power uprate operating parameters and SG tube plugging levels of 0 percent and 10 percent. A 10 percent tube plugging level represents the maximum allowable tube plugging for a single KNPP SG. Analysis showed that pressure differentials were below the design pressure limit of 1800 psi and that ASME Code requirements were met.

The NRC staff finds the licensee evaluation acceptable and, therefore, the staff concludes that the proposed 6.0-percent power uprate will not have a significant impact on SG structural integrity.

3.6.2.4 Tube Vibration, Wear, and Repair Hardware

An analysis was performed to evaluate the potential for increased tube wear resulting from the operation of the SGs in a stretch uprated power condition. The licensee also evaluated the acceptability of various SG tube plug designs for the stretch uprated power operating conditions. Results from the current design wear analysis were modified to account for anticipated changes in secondary side thermal-hydraulic operating conditions due to the stretch uprated power conditions. The calculations performed by the licensee determined that the maximum projected increase in tube wear that could occur was from 3 mils to approximately 4 mils in a cycle at the stretch power uprate condition. Any increase in wear would progress over many cycles and would be observed during routine eddy current inspections.

The NRC staff finds the licensee evaluation acceptable since the maximum projected increase in wear is small. Therefore, any additional wear that could challenge tube integrity would occur over many cycles and would be detected during routine inspection.

The KNPP SGs were replaced in the fall of 2001. Even though the SGs have not been in operation for a long period, the licensee performed an analysis to qualify Westinghouse field-installed weld plugs and the Westinghouse ribbed mechanical plugs. The licensee also performed an analysis to evaluate the acceptability of a tube undercut.

Structural evaluations were performed for both the mechanical and weld plugs for the stretch uprated conditions. Mechanical plug designs included the short and long ribbed original equipment manufacturer plug designs. Although there are no shop welded tube plugs in the KNPP SGs, this design was also evaluated to ensure it is acceptable for the stretch power uprate conditions based on the 1989 ASME Code. Evaluations included the applicable

transients associated with stretch power uprate plus cumulative fatigue design criteria per the ASME Code, Section III. The results from the analyses concluded that both mechanical plug designs satisfy all applicable stress and retention criteria for the stretch power uprate condition. The welded plug calculations for stretch power uprate level operation were also shown to meet the allowable ASME Code values for stress and fatigue usage.

Some tube repair circumstances (e.g. removal of a tube plug by drilling/reaming prior to sleeve installation) may result in removal of a portion of the tube and weld metal. Therefore, the licensee performed an analysis to evaluate the acceptability of a tube with 40 percent undercut, (i.e., removal of 40 percent tube/weld), operating at a 6.0-percent stretch power uprate level condition. Results from the analysis showed that all stresses and fatigue values were within the ASME Code allowable values.

The NRC staff finds the licensee’s evaluation acceptable since the analysis showed a tube in this condition meets ASME Code limits and does not exceed the technical specification tube repair limits.

3.6.2.5 Secondary Side Foreign Object Evaluation

At the time of the stretch power uprate submittals on May 22, 2003, the licensee had not performed the first in-service inspection. A Generic Loose Part Analysis was performed and it assumed a 6.0-percent stretch power uprate level condition. Subsequent to the submittals of the power uprate application, the licensee performed the first in-service inspection and identified some possible loose parts (PLPs). The licensee performed an additional evaluation to specifically address the PLPs found in the inspection. The licensee confirmed that the new evaluation was performed assuming the 6.0-percent stretch power uprate (1773 MWt) and the parameters used in the evaluation were that of the loose parts found.

The NRC staff considers the licensee’s action acceptable since the licensee has evaluated the effects of leaving known loose parts in service and performs routine inspections that would detect the presence of more loose parts in the future.

3.6.2.6 Regulatory Guide 1.121 Analysis

RG 1.121, “Bases for Plugging Degraded [presssurized water reactor] PWR Steam Generator Tubes,” describes an acceptable method for establishing the limiting safe condition of degradation in the tubes, beyond which, tubes found defective by the established in-service inspection shall be removed from service. The level of acceptable degradation is referred to as the repair limit. The allowable tube repair limit, in accordance with RG 1.121, is obtained by incorporating into the resulting structural limit an allowance for continued growth of the flaw and an allowance for eddy current measurement uncertainty. In terms of the stretch power uprate, the structural limit and corrosion rate are affected by parameters such as temperature change and differential pressure (e.g., a change in temperature affects the corrosion rate).

The licensee’s tube structural limits are defined in a Westinghouse topical report, WCAP-15325, assuming a uniform thinning mode of degradation in both the axial and circumferential directions. The assumption of uniform thinning is generally regarded to result in a conservative structural limit for all flaw types occurring in the field. A revised analysis was performed to document applicable tube structural limits for the stretch power uprate conditions.

Although the primary-to-secondary pressure gradients are increased for the stretch power uprate conditions, the analysis showed that the changes were not significant enough to result in an appreciable change to the structural limits. Therefore, the licensee concluded that the existing plugging limit contained in the TSs are adequate.

The NRC staff finds the licensee's evaluation acceptable because it follows the guidance in RG 1.121.

3.6.2.7 Tube Degradation

The potential effects of the 6.0-percent stretch power uprate on SG tube degradation (e.g., axial and/or circumferential stress corrosion cracking, intergranular attack, etc.) were evaluated. The licensee concluded that the 6.0-percent stretch power uprate is not expected to have a significant impact on tube degradation. Degradation resistance is based on the use of the Alloy 690 thermally treated (TT) SG tubing in the KNPP SGs. The Alloy 690 TT is expected to be an improvement over Alloy 600 TT SG tubing which has been shown to be much more resistant to degradation than the original KNPP Alloy 600 mill annealed SG tubing. The licensee's analysis projects very low percentages of tubes plugged at the end of the current license under stretch power uprate operation based upon operating experience with similarly designed SGs, accumulated EFPY of operation, and operating temperature. Therefore, none of the potential degradation mechanisms are significantly affected by the stretch power uprate conditions.

Based on the above rationale, the NRC staff finds the licensee's evaluation to be acceptable.

3.6.2.8 SG Blowdown System

The steam generator blowdown (SGBD) system is used to control the chemical composition and buildup of solids in the SG shell-side (secondary side) water. The SGBD system flow rates are based on water chemistry and tube-sheet sweep (velocity of water across the tube-sheet) required for controlling the buildup of solids. The SGBD system was evaluated in terms of the effect of the stretch power uprate on blowdown flow, design pressure and temperature. The SGBD system design pressure, temperature and flow at the stretch power uprate conditions are all bounded by the system's design parameters.

The inlet pressure to the SGBD system varies with the SG operating pressure. As the pressure decreases, the SGBD system control valves have to open to maintain the desired SGBD system flow rate. The licensee's SGBD system is able to accommodate these SGBD flow rates at stretch power uprate conditions. Therefore, the licensee determined that the stretch power uprate will not impact the system's ability to meet the required flow rates.

The NRC staff finds the licensee evaluation acceptable since new temperature, pressure and flow caused by the proposed 6.0-percent stretch power uprate are bounded by the system's design parameters.

3.6.2.9 Nuclear Steam Supply System Piping

The NRC staff previously approved an amendment for a 1.4 percent MUR on July 8, 2003 (ADAMS Accession No. ML031530734). Under Section 5.5 of the May 22, 2003, application,

the licensee indicated that the maximum RCL piping stresses for the RCL piping and the corresponding code-allowable stress values were combined in accordance with the methods and code criteria as described in WCAP-7840, “Structural Analysis of Reactor Coolant Loop/Support System for NSP (Prairie Island) and WPS (Kewaunee) Nuclear Power Plants, Report No. D 103, February 1972.” This comparison is shown under Table 5.5-2, “RCL Stress Analysis Summary - 7.4 -Percent Power Uprate Program,” which provides a summary of the Hot-Leg, Crossover Leg and Cold-Leg Maximum and Allowable stresses for the following conditions: design stress, upset stress, faulted stress and thermal stress.

The licensee indicated that the RCL piping loads for LBB evaluation for the 7.4-Percent Power Uprate Program were evaluated and found to be acceptable as shown under Subsection 5.5.2 in the May 22, 2003, application. The licensee indicated that Westinghouse performed analyses for the LBB of KNPP primary loop piping in 1987. The results of the analyses were documented in WCAP-11411, Revision 1, “Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Kewaunee, Rev. 1,” and WCAP-11619, “Additional Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Kewaunee.”

The actual results are considered Westinghouse Proprietary Information. The NRC staff review of the actual stresses as provided by the licensee indicate that the actual stresses due to the stretch power uprate are conservatively lower than the allowable stresses and therefore, acceptable for the operating temperatures generated during the 6.0-percent stretch power uprate.

For the LBB evaluation, the licensee used the recommendations and criteria proposed in the “Standard Review Plan for Leak-before-Break Evaluation Procedures,” as stated under Reference 11, page 5-59, of the May 22, 2003, application. The primary loop piping dead weight, normal thermal expansion, Safe Shutdown Earthquake and pressure loads due to the 7.4 percent power uprate program normal operating temperature and pressure were used in the evaluation. The licensee concluded that all the LBB recommended margins were satisfied for the stretch power uprate conditions.

The acceptance criteria under SRP 3.6.3 recommends the following margins:

- Margin of 10 on leak rate.
- Margin of 2.0 on flaw size.
- Margin on loads of 1.0 (absolute summation).

The results of the licensee’s evaluation showed the following:

- Actual margin of 10 exists between the calculated leak rate from the leakage flaw and the leak detection capability of 1 gpm.
- Actual margin of 2.0 or more exists between the critical flaw and the flaw have a leak rate of 10 gpm.
- Actual margin of 1.0 on loads exists.

Based on the actual margins as stated by the licensee, the NRC staff concludes that the LBB acceptance criteria are satisfied for the KNPP NSSS piping at the 6.0-percent stretch power uprate conditions. Furthermore, the temperatures generated during the stretch power uprate

will not exceed the materials' capabilities to withstand service condition stresses. Based on the discussion above, the NRC staff concludes that the NSSS piping materials can withstand the increased temperatures and pressures due to the 6.0-percent stretch power uprate.

3.6.2.10 Balance of Plant Piping

The licensee stated in Section 8.4 of its May 22, 2003, application that an assessment of the BOP piping and supports (including main steam, condensate, feedwater, auxiliary feedwater and SG blowdown systems) piping was performed for a power uprate of 1772 MWt. The licensee concluded that the piping and pipe supports remain in compliance with the USAS B31.1, "Power Piping Code," and that the existing main steam piping remains acceptable for the power uprate condition which are based on the results of analyses with the higher flow rate resulting from the power uprate. In response to the NRC staff's request for additional information (RAI), the licensee provided Table 1, "Maximum Pipe Stress Levels and Allowables for Main Steam." The table indicated that the maximum upset stress condition due to the stretch power uprate was 20,665 pounds per square inch (psi), which did not exceed the allowable stress limit of 21,000 psi. Similarly, the table indicated that the maximum faulted stress condition due to the stretch power uprate was 23,837 psi, which did not exceed the allowable stress limit of 31,500 psi.

The licensee addressed piping system limits due to the stretch power uprate in its response to the RAI under the Table, "Summary of Pipe Stress Levels," for Condensate, Feedwater, Bleed Steam and Heater Drains," shown below.

Piping System	Loading Condition	Existing Stress (psi)	Power Uprate Stress (psi)	Allowable Stress (psi)	Comments
Condensate	Sustained + Thermal	27,986	28,434	37,500	Note 1
Feedwater	Thermal	12,403	12,527	22,500	Note 2
Bleed Steam	Thermal	18,065	18,246	22,500	Note 3
Heater Drains	Thermal	14,907	15,354	22,500	Note 4

Note 1
The stress data shown is for the condensate piping running between Heater 14A&B and the FW pumps which experiences a temperature increase from 360 °F to 366 °F.

Note 2
The stress data shown is for the outside containment feedwater piping located downstream of heaters 15A&B which experiences a temperature increase from 432 °F to 437 °F.

Note 3
The stress data shown is for the bleed steam piping running from the HP turbine to heaters 14A&B which experiences a temperature increase from 365 °F to 368 °F.

Note 4
The stress data shown is for the heater drain piping running from heaters 14A&B to the heater drain tank which experiences a temperature increase from 360 °F to 368 °F.

Based on the information provided by the licensee, it showed that the actual stresses do not exceed the code allowable, and therefore, reasonable assurance is provided that the temperature and stress limits will not be exceeded for the KNPP BOP piping at the stretch power uprate conditions. Based on the information provided by the licensee, the NRC staff concludes that the BOP piping materials can withstand the increased temperatures and pressures due to the 6.0-percent stretch power uprate requested by the licensee.

3.6.2.11 Control Rod Drive Mechanism Housings

In its RAI to the licensee, the NRC staff requested that the licensee discuss its determination made for service adequacy of the materials in the RPV CRDM taking into consideration NRC Bulletins 2002-01 and 2002-02. Both bulletins addressed the generic issue of Primary Water Stress Corrosion Cracking (PWSCC) in the CRDM housings, which affects both the RCPB integrity and RPV head and nozzle degradation.

The licensee's November 5, 2003, RAI response was that Attachment 4, Sections 5.1 and 5.4 provide the structural evaluations of the reactor vessel (including the RPV head) and the CRDMs for the power uprate operating conditions. As stated by the licensee, these structural evaluations performed in support of the stretch power uprate confirmed that the impact of the Performance Capability Working Group design operational parameters, and the NSSS design transients for the power uprate program, are bounded by the parameters and transients for either the RSG Program or the original licensing design analysis. Secondly, the licensee is planning to replace the KNPP reactor vessel head, tentatively scheduled for the fall 2004 outage. The replacement reactor vessel head will be installed with penetrations made of Alloy 690 tubing and Alloy 52 weld metal. If, due to an unforeseen schedule or technical difficulties, the RPV head is not replaced during the fall 2004 outage, the existing head will be in the high susceptibility category of NRC Order EA-03-009. The licensee stated that it would perform all required nondestructive examinations outlined in the subject Order prior to placing the existing RPV head back in service.

Published data indicates that RPV heads constructed using CRDM Alloy 690 tubing with Alloy 52 welds is more resistant to PWSCC, and the NRC staff currently considers these materials an

acceptable material alternative to current construction use of Alloy 600 tubing with Alloy 82 welds. If the RPV head is not replaced per the schedule provided by the licensee, the licensee's alternative to nondestructively test the RPV head per the requirements of Order EA-03-009 provides reasonable assurance of the continued structural integrity of the RPV head.

Based on the commitment by the licensee to either replace the RPV head with superior material or nondestructively test the CRDM nozzle welds per the requirements of NRC Order EA-03-009, the NRC staff concludes that the materials in the CRDM will be acceptable for the service conditions generated under the licensee's 6.0-percent stretch power uprate.

3.6.3 Conclusion

The NRC staff has reviewed the licensee's evaluation of the impact of the proposed stretch power uprate on reactor vessel integrity, SG tube integrity, and FAC programs. The technical areas reviewed by the NRC staff are those discussed in Section 3.6.1 of this SE. Based on the above, the NRC staff concludes that the licensee has adequately addressed these impacts and has demonstrated that the plant will continue to meet the applicable requirements following implementation of the proposed stretch power uprate. Therefore, the NRC staff finds the proposed 6.0-percent stretch power uprate acceptable with respect to materials and chemical engineering.

3.7 <u>Human Factors</u>

3.7.1 Regulatory Evaluation

The area of human factors deals with programs, procedures, training, and plant design features related to operator performance during normal and accident conditions. The NRC staff's human factors evaluation is conducted to confirm that operator performance will not be adversely affected as a result of system changes required for the proposed stretch power uprate. The NRC staff's review covers licensee's plans for addressing changes to operator actions, human-system interfaces, and procedures and training required for the proposed stretch power uprate. The NRC's acceptance criteria for human factors are based on 10 CFR 50.54(i) and (m), 10 CFR 50.120, 10 CFR 55.59, and GDC-19.

3.7.2 Technical Evaluation

The NRC staff has reviewed the following human factors area: (1) operator actions, (2) emergency and abnormal operating procedures, (3) control room controls, displays, and alarms, (4) safety parameter display system, and operator training program and the control room simulator. The licensee has addressed these five areas in its May 22, 2003, application. Following is a summary of the licensee's responses and the NRC staff's conclusions.

3.7.2.1 Operator Actions

The licensee indicated that the proposed stretch power uprate is not expected to have any significant affect on the manner in which the operators control the plant during normal operations or transient conditions. The licensee also indicated that any emergency or abnormal operating procedures regarding AFW TS requirements will be changed prior to implementation

of the stretch power uprate; the licensee made commitment number 12 in Attachment 7 of the May 22, 2003, application to perform the changes. The NRC staff finds the implementation of the proposed stretch power uprate at KNPP will not have an adverse effect either on operator actions or safe operation of the facility.

3.7.2.2 Emergency and Abnormal Operating Procedures

The licensee indicated that there are minor changes needed to the EOPs and Abnormal Operating Procedures (AOPs) as a result of the 6.0-percent stretch power uprate. The licensee stated that the changes would include setpoint changes. The EOPs and AOPs that are entered due to a LONF event will be changed to reflect the new TS requirements for the AFW system. The licensee made commitment number 12 in Attachment 7 of the May 22, 2003, application to perform the changes. Based on the above, the NRC staff finds that necessary procedures will be changed or updated prior to the implementation of the license and TSs changes associated with the proposed stretch power uprate. The NRC staff finds this acceptable.

3.7.2.3 Control Room Controls, Displays, and Alarms

The Ultrasonic Flow Measurement Device (UFMD) at KNPP consists of an Ultrasonic Flow Measurement (UFM) system called “Crossflow” and an ultrasonic temperature measurement (UTM) system called “CORRTEMP.”

The licensee stated that the following will be updated consistent with operation at uprated power 1772 MWt (see Attachment 7, “List of Regulatory Commitments,” of the May 22, 2003, application):

- The plant process computer screen (PPCS), which displays UFMD/UTM correction factors, UFMD OPERATING LIMIT, and [Reactor Thermal Output] RTO OPERATING LIMIT,
- The CROSSFLOW computer program constants,
- The alarm setpoints associated with the UFMD operational limits,
- The UFMD abnormal and alarm procedures as necessary,
- The PPCS computer constants and embedded values,
- Computer constants, such as full power delta T and licensed rated power level, and
- Programs, such as RTO program.

The above updates will be finalized prior to implementing the proposed stretch power uprate according to Commitment No. 14 of Attachment 7, “List of Regulatory Commitments,” of the May 22, 2003, application. The NRC staff finds this acceptable.

The licensee stated that there are no changes to the safety parameter display system as a result of the stretch power uprate. The NRC staff finds this acceptable.

3.7.2.4 Control Room Plant Reference Simulator

The KNPP Simulator Certification was submitted in a letter from C. A. Schrock, Wisconsin Public Service Corporation, to NRC Document Control Desk, dated January 27, 1992. The KNPP simulator will be modified to provide the same information and annunciation that is being changed in the control room. The modifications to the control room simulator will be done in

accordance with the appropriate licensee's site design change procedures. Modifications associated with the stretch power uprate will be completed prior to implementation (See Attachment 7, "List of Regulatory Commitments," of the May 22, 2003, application). This will be finalized prior to implementing the proposed stretch power uprate. The NRC staff finds this acceptable.

3.7.2.5 Operator Training Program

The licensee stated that the operator training program will review the changes made to the TSs and procedures as a result of the stretch power uprate. The licensee's operations training department will determine the extent of training required based on the changes prior to the stretch power uprate implementation. Specific training will be performed associated with the plant procedure changes as determined by the KNPP operations department in accordance with the appropriate plant processes. The licensee will provide appropriate training to the necessary plant staff for changes associated with the implementation of the new rated power (See Attachment 7, "List of Regulatory Commitments," of the May 22, 2003, application). This will be finalized prior to implementing the proposed stretch power uprate. The NRC staff finds this acceptable.

3.7.3 Conclusion

The NRC staff has reviewed the licensee's planned actions related to the human factors area, and concludes that the licensee has adequately considered the impact of the proposed stretch power uprate on changes to operator actions, procedures, plant hardware, and associated training programs to ensure that operators' performance is not adversely affected by the proposed stretch power uprate. The NRC staff further concludes that the licensee will continue to meet the requirements of 10 CFR 50.54(i) and (m), 10 CFR 50.59, 10 CFR 50.120, and 10 CFR 55.59 following implementation of the proposed stretch power uprate. Therefore, the NRC staff finds the proposed 6 percent stretch power uprate acceptable with respect to the human factors aspects of required system changes.

3.8 Plant Systems

3.8.1 Regulatory Evaluation

The NRC staff review in the area of plant systems covers the impact of the proposed stretch power uprate on (1) containment performance analyses and containment systems, (2) safe shutdown fire analyses and required systems, (3) spent fuel pool cooling analyses and systems, (4) flooding analyses, (5) NSSS interface systems, (6) radioactive waste systems, (7) ESF HVAC systems, and (8) safety-related cooling water systems. The review is conducted to verify that the licensee's analyses bound the proposed plant operation at the stretch power level, and that the results of licensee analyses related to the areas under review continue to meet the applicable acceptance criteria following implementation of the proposed stretch power uprate. Guidance for the NRC staff's review of plant systems is contained in Chapters 3, 6, 9, 10, and 11 of NUREG-0800.

3.8.2 Technical Evaluation

3.8.2.1 Containment Performance Analyses and Containment Systems

3.8.2.1.1 Use of GOTHIC 7.0 Computer Code for Containment Analysis

The NRC previously approved the use, for KNPP, of the GOTHIC 6.0 (Version 6.0a) computer code in a September 10, 2001, letter to the licensee. For the power uprate, the licensee proposed using an updated version of the GOTHIC code designated GOTHIC 7.0 (Version 7.0p2). The licensee requested NRC review of GOTHIC 7.0 in a September 30, 2002, letter to the NRC. A September 29, 2003, letter (ADAMS Accession No. ML032681050) from the NRC to the licensee approved GOTHIC 7.0 with several restrictions on the heat and mass transfer models. The licensee submitted re-analyses of the LOCA and main steamline break accident containment integrity analyses in a November 5, 2003, letter. The version of GOTHIC used by the licensee, with the NRC restrictions included, is now designated GOTHIC 7.1 Patch 1. GOTHIC 7.1 is a revised version of GOTHIC 7.0. The changes from GOTHIC 7.0 to GOTHIC 7.1 are described in Appendix B of the GOTHIC 7.1 manual.[4]l The licensee's November 5, 2003 letter states:

> Other than...[those made in response to the NRC September 29, 2003, letter evaluating GOTHIC 7.0] none of the changes implemented in Version 7.1 Patch 1 [including the change from GOTHIC 7.0 to GOTHIC 7.1] will have any impact on the results of the analyses previously submitted for KNPP containment integrity analyses (CIA)...
>
> To confirm that the code differences between the GOTHIC 7.0p2 and 7.1 Patch 1 have no significant unintended impact on the CIA, the original 7.0p2 containment evaluation models were re-run using version 7.1 Patch 1 without making any model changes. Results show no impact on the KNPP CIA.

The NRC staff has examined the changes listed in Appendix B, Table B.1.4, of the GOTHIC 7.1 manual as Changes Which May Affect Results from Previous Versions, and agrees that these changes should not affect the KNPP analyses.

The licensee's November 5, 2003, letter describes the changes made to GOTHIC 7.0 in response to the NRC's September 29, 2003, letter. This description is consistent with the NRC staff's restrictions. Therefore, the use of GOTHIC 7.1 Patch 1 is acceptable for the KNPP 6.0-percent stretch power uprate containment integrity and environmental qualification calculations.

The licensee has used GOTHIC 7.0 without the Mist Diffusion Layer Model (MDLM) option for the power uprate HELB compartment analyses. The MDLM option was the subject of the NRC's restrictions on the use of GOTHIC 7.0p2. The licensee's November 5, 2003, letter states that since the outside containment power uprate analyses do not use the MDLM option for the HELB compartment analyses, these analyses, using the GOTHIC 7.0 or GOTHIC 7.1 code versions comply fully with the NRC's September 29, 2003, letter approving GOTHIC 7.0. The NRC staff agrees.

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[4] GOTHIC Containment Analysis Package User Manual Version 7.1 NAI 8907 Revision 14, January 2003.

3.8.2.1.2 Loss-of-Coolant Accident Containment Analyses

The licensee divides the analysis of the response of the containment to a LOCA into a short-term response and a long-term response. The short-term analyses are also called subcompartment analyses. SRP Section 6.2.1.2 defines a subcompartment as any fully or partially enclosed volume within the primary containment that houses high-energy piping and would limit the flow of fluid to the main containment volume in the event of a rupture of the high-energy piping within the volume. These analyses are performed in order to demonstrate that the walls of subcompartments within the containment will maintain their structural integrity when exposed to the large pressure difference which arises due to rapid pressurization of the fully or partially enclosed volume. The analyses are typically analyzed for a very short time period; the KNPP calculations are done for a time period of 3 seconds following the postulated piping break. The long-term analyses are termed containment integrity analyses and must demonstrate that the design temperature and pressure limits of the containment are not exceeded.

Both the short-term and the long-term containment analyses consist of a calculation of the amount of mass and energy resulting from the pipe break entering the enclosure (subcompartment or containment) and the response of the enclosure to this release of mass and energy.

Due to the 6.0-percent stretch power uprate, the mass and energy release rates and the effect on the containment must be re-evaluated.

3.8.2.1.2.1 Long-Term LOCA Mass and Energy Release

The licensee has calculated the long-term LOCA mass and energy releases to containment using the NRC-approved Westinghouse March 1979 model[5]. Although the licensee has revised the long-term LOCA analyses in accordance with the findings of the NRC's review of the GOTHIC 7.0 containment computer code (see Section 3.8.2.1.1 of this SER input), the mass and energy calculations reported in WCAP 16040-P remain unaffected.

The calculations considered two break locations: the hot-leg and the cold-leg pump suction. WCAP 16040-P states that the double ended hot-leg (DEHL) break has been shown to result in the highest mass and energy release rates during blowdown. This break also results in the fastest reflood rate which increases the rate of release of energy to the containment. However, for the DEHL break, the energy transferred from the steam generators is minimal which results in a reduction in the energy released after blowdown in comparison with the cold-leg break or the pump suction break. The pump suction break includes a relatively high core flooding rate with the transfer of significant heat from the steam generators. Thus, both break locations, the hot-leg and the pump suction, must be considered.

The calculations also consider cases of maximum and minimum safeguards. These are defined in WCAP 16040-P. The minimum safeguards case is the result of an assumed failure of one EDG which results in the loss of one train of safeguards equipment. The maximum

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5 WCAP-10325-P-A (Proprietary), WCAP 10326-A (Non-Proprietary), Westinghouse LOCA Mass and Energy Release Model for Containment Design-March 1979 Version, May 1983.

safeguards case assumes no failure which would affect the amount of ECCS flow. The single-failure for the maximum safeguards case is failure of a train of containment spray.

Section 6.4.1.1.1.1 of WCAP 16040-P describes the input assumptions used for the LOCA mass and energy release calculations. Conservative assumptions are made for the RCS temperature and pressure. According to WCAP 16040-P, the RCS temperatures are chosen to bound the highest average coolant temperature range of all operating cases. A temperature uncertainty of +6.0 of is then added to account for instrument error and deadband. The pressure is based on the nominal pressure plus an uncertainty of +50.1 psi which also accounts for instrument uncertainty and deadband.

RCS conditions are selected by the licensee to maximize the core stored energy at the beginning of the accident and permit its rapid transfer to the coolant (and subsequently to the containment) after accident initiation.

The core decay heat is determined by ANSI/ANS (American National Standards Institute/American Nuclear Society) Standard 5.1-1979[6] with a 2σ uncertainty added. In addition, the fuel stored energy is increased by 15 percent to account for possible variations from nominal fuel design. The rated thermal power level is increased by 0.6 percent to account for measurement uncertainty.

Assumptions are made to overestimate the RCS inventory since a larger inventory results in more reactor coolant discharged into the containment. The nominal reactor coolant system volume is increased by 3 percent. The SG tube plugging percentage is zero. This both maximizes the reactor coolant inventory and the SG heat transfer area which increases the heat transferred to the coolant and therefore, to the containment.

For the mass and energy release calculations, the containment pressure is assumed to be equal to the containment design pressure. This maximizes the rate of reflooding the core and therefore maximizes the heat released to the containment during the reflooding portion of the analysis.

The release of mass to the containment from the postulated break is determined during the blowdown portion of the accident by a critical mass flux correlation. WCAP-10325-P-A utilizes a conservative approach to calculate both the critical mass flux break flow and the break flow rate for the other phases of the accident.

The refill phase of the LOCA, which is the period of time during which the lower reactor vessel plenum is being filled, is neglected since this results in a faster release of mass and energy to the containment.

The mass and energy release evaluation model (WCAP-10325-P-A) consists of several computer codes which model the different phases of the LOCA: blowdown, reflood and post-reflood. These codes are SATAN VI (blowdown), WREFLOOD (reflood), FROTH and EPITOME (post-reflood). The use of these codes was approved as part of the review of WCAP 10325-P-A. The licensee's December 15, 2003, letter, states that although the EPITOME code

[6] ANSI/ANS-5.1 American National Standard for Decay Heat Power in Light-Water Reactors, August 1979.

is not called out by name in either WCAP 10325-P-A or WCAP 8264-P-A[7], the function of the code is described in WCAP 10325-P-A.

Since the LOCA mass and energy release calculations have been performed using methods previously approved by the NRC staff, and since the input assumptions and single-failure assumptions have been made to overestimate the actual mass and energy release, the NRC staff finds the licensee's LOCA mass and energy release calculations to be acceptable.

3.8.2.1.2.2 Long-term LOCA Containment Analysis

The licensee performed analyses to ensure that the design containment pressure and temperature limits are not exceeded at the stretch power uprate conditions. These calculations use the results of the mass and energy release calculations discussed above. As discussed above, the containment calculations are done using the GOTHIC 7.1 Patch 1 computer code[8].

The licensee has used conservative assumptions in calculating the LOCA peak pressure and temperature. These assumptions are described in Section 6.4.1.2 of WCAP 16040-P. The initial values of the containment pressure, temperature and relative humidity are chosen to overestimate the peak containment pressure and temperature. Values of the RWST temperature and service water temperature, which affect the efficiency of the containment spray and the containment fan coil units, respectively, are also chosen to result in a conservatively high containment peak pressure and temperature.

The structural heat sinks are described in Table 2 of the licensee's November 5, 2003, letter. These have been revised from the values previously approved by the NRC staff. The surface areas and thicknesses of the conduits and cable trays are increased to be consistent with more recent plant inventory data.

The maximum safeguards and minimum safeguards cases are both analyzed to determine the worst single-failure.

The results of these analyses are given in Table 5 of the November 5, 2003, letter. The maximum containment pressure of 44.6 psig at 19.9 seconds is a result of the double ended hot-leg break LOCA. Since this case only considers the blowdown phase of the accident, no single failure is considered since the safeguards equipment has not yet actuated. This case also results in the highest containment temperature, 265.0 °F at 19.8 seconds.

The peak pressure after 24 hours is a result of the double ended pump suction break with minimum safeguards. Since it is less than half the initial peak pressure, it is acceptable for the licensee to assume that the containment leakage after 24 hours is half the initial value for calculating offsite doses in accordance with SRP Plan Section 6.2.1.1.II.b. For this case, the licensee assumed different service water temperatures for different periods of time. For the first 24 hours, a service water temperature of 80 °F was assumed. For the period 24 to 168

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7 WCAP 8264-P-A, "Westinghouse Mass and Energy Release Data for Containment Design," August 1975 (an earlier NRC-approved Westinghouse topical report which describes methods for calculating the mass and energy release from a LOCA for containment calculations).

8 GOTHIC Containment Analysis Package User Manual Version 7.1 NAI 8907 Revision, 14 January 2003.

hours, a service water temperature of 73 °F is assumed, and for the remainder of the time a service water temperature of 70 °F is used. Since the average temperature would be expected to be less as the time period increases, this is acceptable.

Since the containment analysis has been performed with acceptable methods and has used conservative assumptions, and since the resulting peak pressure and temperature are less than the design limits, the LOCA containment analyses are acceptable at stretch power uprate conditions.

3.8.2.1.2.3 Short-Term LOCA Mass and Energy Release

The only factor changed by the stretch power uprate which affects the short-term LOCA mass and energy release is the temperature of the reactor coolant. The table below, which is the same as Table 6.4-19 of WCAP 16040-P, shows that the hot-leg, cold-leg and average temperature, T_{avg}, are greater for the stretch power uprate than the current values. Since the short-term releases are controlled by density effects, the lower temperatures of the current licensed operating conditions are more limiting since this results in more mass being injected into the subcompartment. Even though the temperature of the stretch power uprate coolant is higher in the short-term, the effects of the higher energy are not as significant as the higher mass injected in the current analysis.

Comparison of RCS Conditions for Short-Term LOCA Mass and Energy Release		
	RCS Temperature (°F)	
	Current T_{avg}	Power Uprate T_{avg}
Hot Leg	586.3	590.8
Cold Leg	521.9	521.9
T_{avg}	554.1	556.3

The licensee also states that in addition to the higher mass injection of the current analysis, the NRC has previously approved the application of LBB methods for KNPP[9]. LBB eliminates the need to assume the rupture of certain pipes in the reactor coolant system. This is applicable to subcompartment analysis, as discussed in the October 27, 1987, *Federal Register*, which reports the revision of GDC 4 to include LBB[10]. For KNPP, this means that the breaks currently considered for short-term LOCA no longer need to be considered. These breaks include the double ended circumferential rupture of the reactor coolant cold-leg break for the SG

[9] References 2, 3, 4 and 5 of Section 6.4.4 of WCAP 16040-P provide the licensee and NRC communications leading up to and approving LBB for the KNPP. The final approval is provided in a letter to D.C. Hintz, Wisconsin Public Service Corporation, from J. G. Gitter, USNRC, dated March 18, 1988.

[10] 10 CFR Part 50 Modification of General Design Criterion 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures, *Federal Register*, Volume 52, No. 207, October 27, 1987.

compartments, and a reactor vessel inlet break for the reactor cavity region. WCAP 16040-P states that the remaining breaks produce a significantly smaller loading on the structures than the larger breaks which have been eliminated by LBB. Therefore, the licensee has not performed a short-term LOCA analysis. Since the current short-term LOCA analysis is more limiting, and LBB assumptions eliminate the largest breaks from consideration, the NRC staff finds the licensee's proposal to not reanalyze the short-term LOCA effects to be acceptable. The current KNPP licensing basis remains bounding.

3.8.2.1.3 Steamline Break Accident Analysis

The design-basis MSLB accident also can produce significant pressurization of containment. It is not possible to predict which set of conditions will result in the highest containment pressure. Therefore, the licensee has performed a series of analyses assuming different power levels, break sizes, availability of offsite power, and single-failures. The input parameters and assumptions used by the licensee are described in Section 6.4.2.1.2 of WCAP 16040-P.

Power levels of 0 percent, 30 percent, 70 percent and 102 percent of the uprated thermal power were assumed. Based on previous experience, this set of power levels is appropriate and acceptable.

Break sizes of 0.1 ft^2, 0.5 ft^2, 0.8 ft^2, 1.1 ft^2, and 1.4 ft^2 were analyzed. The maximum break size is 1.4 ft^2. The break size is limited to this value by 16-inch diameter flow restrictors integral to the steam generator outlet nozzles. Main steam isolation valves (MSIVs) and downstream non-return check valves ensure that no more than one SG completely blows down as a result of a MSLB.

The licensee examined three single-failures: (1) failure of one feedwater regulating valve to isolate, (2) failure of one MSIV to isolate, and (3) failure of one containment safeguards train to actuate. A safeguards train consists of one train of containment spray and two containment fan coil units. These single-failures are typical of those chosen for this analysis and are acceptable.

The analyses were performed assuming loss of offsite power and offsite power available. The analyses calculate the quality of the steam exiting the break. For the intact SGs, the flow before isolation is assumed saturated steam (quality equal to one).

NRC Bulletin 80-04 requested that licensees consider the consequences of continued feedwater flow. The licensee stated (Section 6.4.2.1.2) that the feedwater was assumed to trip on the SI signal (with a conservative coastdown) and the condensate pumps were assumed to continue running throughout the accident (for the case in which offsite power is not lost). The AFW pumps operate until terminated by the operator at 600 seconds. Note that the limiting case for the 6.0-percent stretch power uprate is a steamline break at 0 percent power. At this power level (hot zero power), the main feedwater system is not in use.

The same initial conditions, representation of the structural heat sinks and containment fan coil unit performance characteristics were used for both the LOCA and the MSLB accident with several exceptions for the MSLB, as discussed in Section 3.8.2.1.3.2.

3.8.2.1.3.1 MSLB Mass and Energy Release Calculations

The methods used to calculate the mass and energy release for the MSLB accident were approved by the NRC staff in a SE accompanying an NRC September 10, 2001, letter to the licensee.[11] The RETRAN 3D code is used for these calculations as described in the September 10, 2001, letter and the references to licensee documents therein. These methods remain applicable and are acceptable for stretch power uprate conditions.

Table 6 of the licensee's November 5, 2003, letter, provides the revised results of the KNPP MSLB analyses mass and energy release calculations. Since these results were obtained with acceptable computer codes and used conservative, acceptable assumptions for the input, as discussed above, the NRC staff finds the licensee's 6.0-percent stretch power uprate MSLB mass and energy calculations to be acceptable.

3.8.2.1.3.2 MSLB Containment Calculations

The MSLB accident calculations are performed using the GOTHIC code (GOTHIC 7.1 Patch 1), as discussed above of this SER input.

In order to compensate, in part, for the conditions placed on the use of GOTHIC by the NRC's September 29, 2003, SER discussed in Section 3.8.2.1.1 of this SER input, the licensee has proposed the use of several additional heat sinks not included in the original (May 22, 2003) power uprate MSLB accident containment analyses. These include credit for the ECCS accumulators as heat sinks and modeling heat transfer to the shield building. During a MSLB accident, the licensee states that the accumulator temperature is below that of the containment atmosphere. The shield building is modeled as an enclosed volume initially at one atmosphere pressure and 120 °F. Credit for these additional heat sinks is acceptable since they would always be available for removing heat from the containment atmosphere during the accident.

Table 7 of the licensee's November 5, 2003, letter, provides the results of the KNPP revised MSLB analyses. The peak pressure results from a 1.4 ft^2 break at 0 percent power. Offsite power is available and a single-failure of one train of safeguards is assumed. The peak temperature results from a 1.4 ft^2 break at 30 percent power. The peak pressure is calculated to be 45.68 psig and the peak temperature is calculated to be 266.6 °F. Both are lower than the respective design limits and are therefore, acceptable.

Since these results were performed with an acceptable computer code and used conservative, acceptable assumptions for the input, the NRC staff finds the licensee's 6.0-percent stretch power uprate containment integrity analyses to be acceptable.

3.8.2.1.4 HELB Outside Containment

Three sections of WCAP 16040-P deal with the consequences of HELBs outside containment. These three sections of WCAP 16040-P evaluate the conditions (pressure and temperature) in the room or compartment associated with the pipe break. Section 6.5.2, "Main Steamline Break Outside Containment Response Analysis," of WCAP 16040-P discusses the response to a

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[11] Letter from J. G. Lamb, USNRC, to M.E. Reddeman, NMC, September 10, 2001.

postulated steamline break of regions within the auxiliary building. Sections 8.5, “Structural Evaluation,” of WCAP 16040-P deals with the loads due to pressure differentials across walls caused by the increase in pressure caused by the HELB in a room. Section 8.9, “Equipment Qualification,” of WCAP 16040-P evaluates the pressure and temperature conditions in a room due to a HELB in that room.

The licensee states that it is not necessary to reanalyze the structural analysis for HELBs for the auxiliary building regions since the worst-case condition is zero power and these conditions remain unaffected by the power uprate. The NRC staff concurs and the licensee’s HELB analyses for structural analyses remain acceptable.

The stretch power uprate resulted in revised PT profiles for the auxiliary building for the EQ envelopes. The mass and energy input for these calculations was determined with the Westinghouse LOFTRAN computer code[12] [13]. This code has been previously approved by the NRC for calculating mass and energy releases from a ruptured steamline and their application to KNPP is acceptable.

The GOTHIC 7.0 computer code was used to determine the PT profiles. The mass and energy releases assumed no water entrainment for the entire break spectrum. This assumption maximizes the temperature of the equipment for environmental qualification. Attachment 4 of the licensee’s November 5, 2003, letter, states that a GOTHIC (Version 6.0) auxiliary building compartment model was developed and used for the KNPP replacement SG HELB analyses. The GOTHIC 7.0 code version was benchmarked to the GOTHIC 6.0 replacement SG results with excellent agreement. The KNPP GOTHIC 7.0 auxiliary building model was applied to the stretch power uprate HELB analyses. The MDLM option, the use of which was restricted by the NRC’s September 29, 2003, GOTHIC 7.0 SER, was not used for these calculations. Therefore, the use of GOTHIC 7.0 without the MDLM option, or the use of GOTHIC 7.1 Patch 1, which is equivalent (see Section 3.8.2.1.1) is acceptable.

Since the environmental calculations were done with approved methods and conservative assumptions, the NRC staff finds the licensee’s EQ calculations to be acceptable.

3.8.2.1.5 Generation and Disposition of Hydrogen

Section 6.4.3 of WCAP 16040-P contains an evaluation of the hydrogen generation in the containment following a LOCA, performed at stretch power uprate conditions. This evaluation is based on parameters and assumptions valid for the stretch power uprate and used Westinghouse methods and the guidance of RG 1.7[14].

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12 WCAP 7907-P-A, “LOFTRAN Code Description,” Westinghouse Electric Corporation, April 1984.

13 WCAP-8822 (Proprietary) and WCAP-8860 (Non-Proprietary), “Mass and energy Releases Following a steam line Rupture,” September 1976; WCAP-8822-S1-P-A (Proprietary) and WCAP-8860-S1-A (Non-Proprietary), Supplement 1 - Calculations of Steam Superheat in mass/Energy Releases Following a Steam Line Rupture, September 1986; WCAP-8822-S2-P-A (Proprietary) and WCAP 8860-S2-A (Non-Proprietary), Supplement 2 - “Impact of Steam Superheat in Mass/Energy Releases Following a Steam-line Rupture for Dry and Subatmospheric Containment Designs,” September 1986.

14 USNRC Regulatory Guide 1.7, ”Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident,” Revision 2, November 1978.

Effective October 16, 2003, the NRC amended its regulations on combustible gas control as published in the *Federal Register* (Volume 68, No. 179, page 54123). The revised rule change is applicable to power reactors. It eliminates the requirements for hydrogen recombiners and hydrogen purge systems. With this change, the licensee's evaluation is beyond that required by the revised rule. KNPP is in compliance with the revised rule.

3.8.2.1.6 HVAC Systems

In response to the NRC staff's RAI regarding the potential impact of the stretch power uprate on those HVACS discussed in the SRP, the licensee stated that the KNPP design preceded the development of the SRP. Nevertheless, the KNPP Auxiliary Building Ventilation System, as discussed in Attachment 4 to the licensee's May 22, 2003, application, has separate ventilation systems for the equipment rooms, spent fuel pool, containment penetration, control room and non-radioactive areas. This one system crosses all the SRP section boundaries referenced in the NRC staff's question. As a result of the licensee's response and the NRC staff's review of Attachment 4 to the licensee's submittals, the NRC staff found that the licensee stated that the stretch power uprate at KNPP will result in an increase in heat loss to the plant environment from main steam, SG blowdown, and feedwater piping. Other systems may experience a slight increase. In addition, the increased electrical loading may also slightly increase cooling requirements. The following HVAC systems were evaluated to ensure that sufficient margin and capability exist to operate satisfactorily in support of the 6.0-percent stretch power uprate:

Auxiliary Building Ventilation and Cooling Systems
Auxiliary Equipment Areas
Spent Fuel Pool Area
Non-Radioactive Area
Control Room Area
Control Room Air Conditioning System
Auxiliary Building Special Ventilation System (Zone SV)
The Turbine Building Ventilation System
Diesel Generator Cooling
Containment Ventilation System
Shield Building Ventilation System

For the NRC staff's review, the licensee provided an HVAC evaluation that determined the impact of the stretch power uprate on the subject systems' ability to maintain normal operating temperatures. The subject systems were evaluated by determining the changes in the operation of equipment located in the areas they serve due to the stretch power uprate.

For the Auxiliary Building Ventilation and Cooling System, the licensee stated that some component electrical loads may increase; however, these increases are bounded by component ratings. Some local area piping temperatures may increase due to the stretch power uprate. The heat increase in the penetration area is proportional the increase in temperature of the fluid system. However, the expected heat load increase on penetration cooling would be less than 2 percent. Operating temperatures of the fluid piped through the auxiliary building will remain bounded by design, with a minor increase in current temperatures.

The results of the licensee's evaluation indicated that based upon operating temperatures, the penetration cooling may experience a 2 percent increase in cooling loads. The penetration

cooling flow is only 10 percent of the auxiliary building ventilation flow, so the increase to the entire system would be only 0.2 percent. An NRC staff assessment of this evaluation finds this acceptable, because, as the licensee indicated, 0.2 percent is bounded by the rated values of the system.

The Spent Fuel Pool Area Ventilation, Turbine Building Ventilation, and Diesel Generator Cooling systems are unaffected by the stretch power uprate, because the capacity of these systems bounds the stretch power uprate loads. The NRC staff agrees with the licensee's assessment and finds this acceptable.

With respect to the Control Room Air Conditioning System, the licensee stated that the control room heat loads do not increase as a result of the stretch power uprate; therefore, the control room HVAC system is not impacted. The NRC staff agrees with the licensee and therefore finds this acceptable. In addition, the NRC staff's judgment is that, if there were an additional heat load beyond the licensee's anticipation, the capacity of the system will bound that potential increase.

For the Containment Vessel Air Handling System, the licensee stated that the temperature increase inside the containment is estimated to increase by 2 percent. This correlates to a 1.2 °F increase, and the highest summer/fall average temperature is estimated to be 112 °F; therefore, the containment temperature limits are not violated. The NRC staff's judgment is that this negligible increase in temperature is within the capacity of the system, and therefore finds the licensee's position acceptable.

For the Shield Building Ventilation System, the licensee stated that the power uprate requirements to produce a negative pressure within the shield annulus following a LOCA are affected by the containment temperature profile for a LOCA. The result is a slight increase in the time required to produce negative pressure in the annulus. The licensee also stated that this increase is bounded by the system and USAR stated required time. The NRC staff finds this acceptable, because the requirement, as specified in the USAR, is not exceeded.

The licensee stated that based on the HVAC system evaluations, the instruments and controls for the systems listed above are not affected by the stretch power uprate. The licensee concluded and the NRC staff agrees that due to negligible changes in environmental conditions, or margin in design, the subject systems' ability to maintain operating temperature at or below the maximum normal operating temperature is not impacted by the 6.0-percent stretch power uprate. Therefore, no modifications are required to the HVAC systems to support the 6.0-percent stretch power uprate.

3.8.2.1.7 Conclusions

The mass and energy release calculations for the LOCA and MSLB are acceptable. They were calculated using methods contained in licensee reports approved by the NRC in a September 10, 2001, letter to the licensee. The reanalysis of the KNPP containment, utilizing the GOTHIC 7.1 Patch 1 computer code, for the LOCA and MSLB accidents, is acceptable. The licensee has used the code in accordance with the NRC recommendations transmitted to the licensee in a September 29, 2003, letter. The methods for analyzing HELBs outside containment for EQ used GOTHIC 7.0 with conservative assumptions. The NRC staff has concluded that for these specific calculations, the use of either GOTHIC 7.0 or GOTHIC 7.1

patch 1 is acceptable. The structural analysis of HELBs in the auxiliary building was not redone for stretch power uprate since the zero power case is limiting. The safety-related HVAC features of KNPP were reviewed against the pertinent criteria and found to be acceptable. The licensee is in compliance with 10 CFR 50.44 with respect to combustible gas control. Since the analytical methods are acceptable and conservative assumptions were made which continue to satisfy the requirements of GDCs 4,16, 50 and 38, the licensee's analyses to support the 6.0-percent stretch power uprate in the areas of containment integrity, subcompartment analysis, HELB outside containment, hydrogen generation and HVAC are acceptable.

3.8.2.2 Safe Shutdown Fire Analyses and Required Systems

The licensee states that KNPP will be capable of meeting the Appendix R Requirement to achieve cold shutdown of the plant within 72 hours following a fire at the stretch power uprate level. Section 4.1.4.3.2, of the May 22, 2003, application, states that the surviving train will be capable of reducing the RCS to cold shutdown conditions within the requirements of Appendix R at stretch power uprate conditions according to the Appendix R cooldown analysis.

In the RAI dated October 7, 2003, the NRC staff requested information regarding changes that the stretch power uprate request may have on the fire protection program and on the post-fire safe shutdown capability in accordance with Appendix R. The licensee responded to the RAI on November 5, 2003, that the stretch power uprate does not involve physical changes to the fire protection program.

In a letter dated December 15, 2003, the licensee provided additional clarification regarding SG dryout times. The licensee verified that the time to SG dryout is greater than the 10 CFR Part 50, Appendix R, design-basis operator response time to respond to the SGs drying out. The calculation concludes that design requirement for response time does not need to be revised for the 6.0-percent stretch power uprate. Based on this calculation, no changes to the Appendix R shutdown procedures are required for the 6.0-percent stretch power uprate.

In a letter dated January 30, 2004, licensee provided additional clarification on using the RWST to borate the RCS to cold shutdown without the availability of letdown. The licensee performed a calculation that demonstrates that the RWST satisfies the plant's need of a boration source with or without letdown. Based on this calculation, no changes to the Appendix R shutdown procedures are required for the 6.0-percent stretch power uprate.

The NRC staff has reviewed the application and the letters dated November 5, December 15, 2003, and January 30, 2004. The NRC staff concludes, based on the technical content of the submittals, that plant operations at the proposed 6.0-percent uprated power level will have an insignificant or no impact on licensee's compliance with their Fire Protection licensing basis, 10 CFR 50.48, or applicable portions of 10 CFR Part 50, Appendix R. Therefor, the NRC staff finds the Appendix R analyses acceptable for the proposed 6.0-percent stretch power uprate.

3.8.2.3 Main Steam System

The function of the Main Steam (MS) System is to transport saturated steam from the steam generators (SGs) to the high-pressure (HP) turbine, over the entire operating range from system warmup to full power operation. The MS system also provides steam for the moisture separator reheaters (MSRs), turbine glands (sealing steam), auxiliary feedwater (AFW) pump

turbine, and other plant auxiliaries. Portions of the MS system pressure boundary from the SGs up to and including the main steam isolation valves (MSIVs), the SG power-operated atmospheric relief valves (ARVs), and the main steam safety valves (MSSVs) are designed as safety-related. The portion of the MS system piping that supplies steam to the AFW pump turbine, including the turbine drive exhaust piping, is also safety-related. The balance of the remaining MS system pressure boundary is non-safety-related (May 22, 2003, application, Attachment 4, Section 8.3.1.1). According to KNPP's USAR (Section 10.2.2), the design pressure of the MS system is 1085 psig at 600 °F. Based on a review of the information that was submitted, the only area of the MS system that was impacted by the proposed stretch power uprate such that reactor safety considerations could be affected involves the ARVs.

At KNPP, there are two ARVs (one in each of the two MS lines) located upstream of the MSIVs. The ARVs are sized to have a capacity of 10 percent of the maximum calculated steam flow. The licensee has evaluated the capability of the ARVs to perform their design-basis functions following power uprate, and has determined that the original ARV design basis in terms of cooldown capability will continue to be maintained. This conclusion was reached based on analysis of the most limiting SG tube rupture (SGTR) event, which was found to be bounding with respect to sizing the ARVs. The NRC staff has reviewed the licensee's assessment of the effects of the proposed stretch power uprate on the ARVs and based on the information that was submitted, the NRC staff finds that the ARVs will continue to be able to satisfy their design-basis cooldown functions following the 6.0-percent stretch power uprate. The NRC staff finds that the main steam system is acceptable for the proposed 6.0-percent stretch power uprate.

3.8.2.4 Condensate and Feedwater Systems

The Condensate and Feedwater (FW) Systems provide makeup water from the condenser to the SGs. The condensate pumps take a suction from the condenser hotwell and discharge the condensate to the suction of the FW pumps via the air ejector condenser, the gland sealing steam condenser, and the low-pressure FW heaters. The FW pumps supply FW through the high-pressure FW heaters, the FW regulating valves (FRVs), and the Main Feedwater Isolation Valves (MFIVs), to the SGs. While the Condensate and FW systems are not safety-related, the MFIVs are safety-related and are relied upon to isolate FW to/from the SGs during certain event scenarios. These systems are described in the KNPP USAR, Section 10.2.2. Based on a review of the information that was submitted, the only areas of the Condensate and FW systems that were impacted by the proposed stretch power uprate such that reactor safety considerations could be affected involves the Condensate and FW Pumps, the FRVs, and the MFIVs.

NMC evaluated the Condensate and FW systems and associated piping, pumps, valves, and pressure-retaining components to confirm their ability to operate at the proposed stretch power uprate conditions. A hydraulic flow model was used to analyze and evaluate the performance of the Condensate and FW systems under the proposed stretch power uprate conditions, considering both normal plant operation and postulated transient conditions. The evaluation was focused on determining the impact of the proposed stretch power uprate on: (1) system pressures and temperatures; (2) operation of the condensate and FW pumps, including flow capacity, discharge pressure, and net positive suction head (NPSH); (3) operation of the FW heaters; and (4) isolation capability afforded by the FRVs and the MFIVs.

Based on its hydraulic flow model, NMC determined that the available NPSH for the FW pumps is not adequate for the 50-percent load rejection transient or for a 50-percent reduction in heater drain flow with normal condensate flow. For both of these scenarios, the licensee determined that the low-pressure FW heaters need to be bypassed to obtain adequate available NPSH for continued operation of the FW pumps. The licensee concluded that the current set point of 220 psig for automatically bypassing the low-pressure FW heater is adequate for assuring sufficient NPSH for the FW pumps for the proposed stretch power uprate conditions. In the letter dated January 30, 2004, Enclosure 2 (Page 10-11, Item 3.C and Table 3.2.3-3b), the licensee provided the current and predicted pump values at stretch power uprate conditions. The information provided indicates that available NPSH is greater than the required NPSH. The licensee also indicated that FW pump operation will be monitored during power ascension to the uprated conditions to confirm that operation remains as expected. The NRC staff finds the licensee’s assessment and continued monitoring during power ascension to be acceptable for assuring adequate normal FW capability for the SGs for the proposed stretch power uprate.

The FRVs and MFIVs are located outside the containment and are designed to isolate FW flow to the SGs following unisolable steam (or feedwater) line breaks or malfunctions in the SG level control system. Isolation of FW flow is required to prevent containment overpressurization and excessive RCS cooldown. The closure requirements imposed on the FRVs and MFIVs may cause large dynamic pressure changes and must be considered in the design of the valves and associated piping. However, the licensee indicated that while FW flow rate requirements will increase slightly for the proposed stretch power uprate condition, FW system design requirements and limiting assumptions for the current licensed power level continue to bound the worst-case valve loads that might be imposed as a result of the proposed stretch power uprate. Therefore, based on the information that was provided, the NRC staff finds that the FRVs and MFIVs are acceptable for the proposed stretch power uprate conditions.

The NRC staff finds that the condensate and FW systems are acceptable for the proposed 6.0-percent stretch power uprate.

3.8.2.5 Auxiliary Feedwater System

The AFW system supplies FW to the secondary side of the SGs when the normal FW supply is not available. The system removes decay heat from the reactor core by heat exchange in the SGs when the main feedwater pump(s) are not functional, and maintains the required heat sink for the RCS. The AFW system also functions as an engineered safeguard system and is directly relied upon to dissipate reactor decay heat and to prevent core damage in the event of transients and accidents, such as during a loss of normal feedwater (LONF) or during a secondary system pipe break. The AFW system consists of three pumps (two motor-driven and one-turbine driven), associated valves and piping, and control systems to enable the AFW system to satisfy single active failure and diversity of power source/type considerations. All three pumps are interconnected on the discharge side by a cross-over pipe with two normally open motor operated isolation valves. Each pump is capable of supplying FW to either or both SGs. The normal (short-term) assured water supply for the AFW pumps is from two 75,000-gallon condensate storage tanks (CSTs). The backup (longer-term) assured water supply for the AFW pumps is provided by a Category I service water (SW) system. The functional details and the flow diagram of the AFW system are provided in KNPP’s USAR (Section 6.6).

To fulfill the engineered safety features (ESFs) design functions, sufficient FW must be available during transient and accident conditions to enable the plant to be placed in a safe shutdown condition. The limiting event for establishing the minimum required CST inventory requirement (assured short-term AFW water supply) is station blackout (SBO). Since KNPP is required to have a 4-hour coping period, the plant licensing basis requires that sufficient useable CST inventory must be available to bring the plant from full power to hot-standby conditions, and to maintain the plant in hot-standby for 4 hours. As explained in Enclosure 2. Page 15 of the letter dated January 30, 2004, the inventory requirements of the CST have evolved over time to the current requirement of 39,000 gallons. According to Attachment 4 of the May 22, 2003, application, the current minimum CST useable inventory will be impacted by the proposed stretch power uprate due to the increased reactor decay heat and revised operating conditions that will exist. Therefore, the licensee performed a new analysis to determine the minimum useable CST inventory that would be required to support the proposed stretch power uprate conditions. The results indicates that the minimum useable inventory in the CSTs should be increased from 39,000 gallons to 41,500 gallons in order to meet the SBO licensing basis for the proposed stretch power uprate conditions. As discussed in Section 4.6 of this Safety Evaluation, the licensee has proposed changes to the KNPP TS inventory requirements for the CSTs so that existing licensing basis requirements will continue to be satisfied following the stretch power uprate. The NRC staff considers the licensee's assessment of CST inventory requirements to be acceptable for the proposed 6.0-percent stretch power uprate.

In addition to CST inventory requirements, the licensee also assessed the capability of the AFW system to provide FW to the SGs during postulated accident and transient conditions following the proposed stretch power uprate. Based on this assessment, the licensee determined that for the most limiting case of a LONF event, flow from two AFW pumps would be required; whereas for the current licensed power level, flow from only one AFW pump is required. The licensee determined that for all other event and transient scenarios, flow from only one AFW pump would be required under the stretch power uprate conditions. As discussed in Section 4.7 of this Safety Evaluation, the licensee has proposed a change to the KNPP TS requirements for the AFW system in order to assure continued decay heat removal capability during stretch power uprate conditions. The NRC staff considers the licensee's assessment of AFW pump capability to be acceptable.

As the SW system is the assured (longer-term) source of water for the AFW system, the NRC staff questioned the capability of the SGs to cool the RCS to the point where the Residual Heat Removal (RHR) System could be placed in service following implementation of the proposed stretch power uprate. The source of water for the SW system is Lake Michigan, which is known to contain various contaminants, such as dissolved and suspended solids, organic materials, silt, and debris. The NRC staff asked that the licensee address the impact that the lake contaminants would have on the heat transfer capability of the SGs (including fouling of heat transfer surfaces and clogging of the moisture separators and steam dryers), and confirm that the SGs will still be able to perform their function when using lake water as the source of FW following the proposed stretch power uprate.

As discussed in the letter dated February 20, 2004, Enclosure 2 (Page 2), the licensee performed an analysis assuming that the amount of water from Lake Michigan that was necessary for SG cooldown of the RCS to RHR entry conditions was 900,000 gallons. This amount of lake water would provide FW to the SGs for 75 hours assuming 200 gpm are

required for cooldown of the RCS. NMC established worst-case (bounding) estimates of the concentrations of contaminants in the Lake Michigan water supply that could exist over a 75-hour period based on chemical analyses of the water and daily trend data that were available. Assuming that all of the contaminants are concentrated in the SGs over the 75-hour assumed mission time, the licensee analyzed the impact that these contaminants would have on SG heat transfer capability and determined that (at most) SG heat transfer efficiency could be reduced by 28.5-percent. The licensee also evaluated the potential effects of the lake water contaminants on the SG moisture separators and air dryers, taking into consideration design details, component sizing, temperature effects, and past experience. The licensee concluded that no significant performance degradation is expected. As an added measure, Westinghouse has reviewed the licensee's assessment of the impact of lake water contaminants on SG performance and concurs with the analytical approach that was used by the licensee and the conclusions that were reached. Based on the information that was provided, the NRC staff agrees that the licensee's assessment appears to be conservative and finds that the licensee has adequately demonstrated that the SGs will be able to cool the RCS to RHR entry conditions using lake water following the proposed 6.0-percent stretch power uprate.

The NRC staff finds that the AFW system is acceptable for the proposed 6.0-percent stretch power uprate.

3.8.2.6 Spent Fuel Pool Cooling System

The function of the Spent Fuel Pool (SFP) cooling system is to remove the decay heat from the spent fuel assemblies stored in the SFP and to maintain the pool water temperature below a maximum temperature of 150 °F during plant operation and refueling (with full core offload), and to maintain its cooling function during and after a seismic event (May 22, 2003, application, Attachment 4, Section 8.3.8.1). The current licensing requirement for KNPP is to maintain the maximum bulk temperature of the SFP water below 150 °F for all scenarios (KNPP - License Amendment No. 150, Page 9 of the staff's Safety Evaluation Report). The current design of SFP cooling system for KNPP is adequate to maintain the SFP water temperature below 150 °F with the maximum number of fuel assemblies stored in the pool. Since the decay heat rate of the spent fuel is a function of the core power level, the proposed stretch power uprate will result in higher heat loads and temperatures for the SFP.

Referring to Section 8.3.8.4 of Attachment 4 to the May 22, 2003, application, to ensure adequate SFP cooling capability following the proposed stretch power uprate, the licensee performed a heat load calculation using conservative assumptions (i.e., a service water temperature of 80 °F, initial pool temperature of 120 °F, 5 percent tube plugging of the SFP cooling system heat exchanger, and full core off-load to the pool at 168 hours after reactor shutdown). The resulting temperature of the SFP was calculated to be approximately 163 °F. The licensee further analyzed that with a loss of the SFP cooling system, the SFP water temperature will rise from 163 °F to the boiling point of 212 °F in about 5.25 hours; whereas, it would take about 6.5 hours for the SFP water to reach 212 °F from 150 °F. For the worst-case scenario, the boil-off rate of the SFP water is 42 gpm.

Because the licensee's analysis indicates that the SFP design-basis temperature criteria could be exceeded as consequence of the proposed stretch power uprate, the licensee has chosen to perform a cycle-specific heat load calculation prior to each refueling outage (see November 5,

2003, letter, RAI responses to Questions 48.a and b). Based on the May 22, 2003, application, Attachment 4, Section 8.3.8.4, the licensee indicated that if the heat load calculation shows that the pool temperature will exceed 150 °F, then movement of fuel from the reactor into the SFP will be delayed until the fuel has decayed to a point where the SFP temperature criteria will not be exceeded. Also, the licensee plans to document the hold time if it goes beyond the TS limit of 148 hours. According to the May 22, 2003, application, Attachment 7, Item 5, the licensee's Reactor Engineering Refueling Procedures have already incorporated this confirmatory calculation as a requirement. The new requirement will administratively control the in-core hold time of the fuel after shutdown to ensure the SFP design-basis temperature limitations are not exceeded.

Thus, the licensee plans to administratively control the in-core hold time of the fuel after reactor shutdown to ensure that the SFP temperature does not exceed 150 °F. Consequently, the licensee has concluded that physical or analytical modifications to the SFP or its cooling system are not necessary in order to accommodate the proposed stretch power uprate conditions. The licensee also indicated that in the case of a total loss of the SFP cooling system, an alternate cooling system will be available. Likewise, if the SFP water begins to boil-off, alternate make-up sources will be made available. The licensee's plan to perform cycle-specific analysis as discussed above is consistent with the NRC staff's review criteria and will assure that design-basis SFP temperature limitations are not exceeded.

The NRC staff has reviewed the information that was provided and the licensee's assessment of the effects of the proposed stretch power uprate on the SFP cooling system. Based on the licensee's assessment and plans to: (1) perform cycle-specific analysis and administratively control the in-core hold time of the fuel after reactor shutdown to ensure that the SFP temperature will not exceed 150 °F, (2) maintain alternate standby cooling capability readily available to mitigate a total loss of the SFP cooling system function, and (3) maintain alternate SFP makeup capability in order to mitigate unexpected boil-off of the SFP as discussed above, the NRC staff finds that the licensee has adequately considered and addressed any adverse impacts that the proposed stretch power uprate may have on the SFP cooling system and makeup capability. Therefore, the SFP cooling system is considered to be capable of performing its licensing-basis functions following the proposed 6.0-percent stretch power uprate.

3.8.2.7 Steam Dump System

The Steam Dump (SD) System provides a steam flow path directly to the condenser and/or atmosphere which bypasses the turbine, thereby providing the capability to accommodate turbine load transients without forcing a reactor trip.

The licensee performed an analysis of the SD system at the proposed stretch power uprate conditions and based on this analysis, determined that the SD system was originally sized to accommodate a steam flow equal to about 85 percent of the maximum calculated full-power steam flow (40 percent condenser dump and 45 percent atmospheric dump) to permit external load reductions up to 100 percent of the rated electric load. For the proposed stretch power uprate, as discussed in the May 22, 2003, application, Attachment 4 (Sections 4.2.4.2 and 4.2.5.2), the maximum required load reduction is being relaxed from 100 percent to 50 percent of the rated electrical load. This reduced load reduction capability results in a corresponding reduction in the maximum required steam dump capacity from 85 percent to 40 percent of the

rated steam flow. In light of this change, the steam dump capacity for the range of NSSS design parameters that are proposed for the stretch power uprate exceeds the minimum recommended capacity of 40 percent of rated steam flow for load reductions up to 50 percent of electrical load and remains acceptable for the proposed stretch power uprate conditions. Based on the licensee's assessment, and recognizing that the SD system is non-safety-related and not credited in the accident analysis for mitigation of transient or accident conditions, the NRC staff finds the proposed reduction in SD system steam flow capacity to be acceptable.

3.8.2.8 Service Water System/Ultimate Heat Sink

The Service Water (SW) System supplies water from Lake Michigan for cooling equipment in the turbine building, the containment fan coil units, and reactor auxiliary systems. Lake Michigan is the Ultimate Heat Sink (November 5, 2003, letter, RAI response to Question 47.a and b). Further, the purpose of the SW system is to provide redundant cooling water supplies for the engineered safeguards systems and equipment during post accident conditions. According to KNPP USAR (Section 9.6.2), the SW system provides a backup water supply to the AFW system when the primary source of water from the CST is not adequately available to the AFW pumps.

As discussed in the May 22, 2003, application, Attachment 4 (Section 8.3.10), the licensee has determined that the required SW flow rates to engineered safeguards equipment for accident conditions are not impacted by the proposed stretch power uprate since the current analysis was based on conditions that remain bounding. The most significant impact of the proposed stretch power uprate on the SW system is a required increase in the turbine building flow requirements for normal full power conditions. The proposed stretch power uprate, in conjunction with an increase to 80 °F in the maximum allowed SW temperature (USAR, Page 9.3.8 and Table 14.3.4-19), results in an increase in the required SW flow to the Turbine Building of about 73-percent above the current requirements with 66 °F SW temperature for normal full power operation. Other SW heat loads do not require any increase in SW flow for normal and accident conditions above those already established for the current power level. Consequently, the licensee has determined that no changes or equipment additions are required for the SW system or for the Ultimate Heat Sink to support the proposed stretch power uprate.

Additionally, as discussed in the letter dated January 30, 2004, Enclosure 2 (Pages 19 and 20), NMC addressed the concerns expressed in Generic Letter (GL) 96-06 with regard to the potential for flashing and two-phase flow in the SW system piping downstream of the Containment Fan Coil Units (CFCU). Based on analysis that was performed for the proposed stretch power uprate condition, the licensee concluded that the two-phase flow conditions for the above piping is bounding and remains valid for stretch power uprate operation. The licensee has also assessed the impact of the proposed stretch power uprate on the GL 96-06 waterhammer evaluation and has determined that the potential for waterhammer is not impacted as well.

Also, as discussed in Section 3.8.2.5 of this Safety Evaluation, the licensee assessed the use of lake water as the assured long-term source of cooling water for the SGs following the proposed stretch power uprate.

The NRC staff has reviewed the information that was provided and the licensee's assessment referred to above regarding the effects of the proposed stretch power uprate on the capability of the SW system to perform its function. Based on a review of the licensee's assessments, the NRC staff finds that the licensee has adequately considered and addressed any adverse impacts that the proposed stretch power uprate may have on the SW system and Ultimate Heat Sink; and the NRC staff agrees and finds acceptable that the SW system and the Ultimate Heat Sink will remain capable of performing their required safety functions following the proposed 6.0-percent stretch power uprate.

3.8.2.9 Bleed Steam System

The Bleed Steam System conveys steam extracted from various stages of the high-pressure (HP) and low-pressure (LP) turbines to the shell side of the FW heaters and to the moisture separator side of the MSRs. Five stages of steam extraction for FW heating are provided; two from high-pressure turbine, one of which is at the exhaust, and three stages from the LP turbines. The FW heaters for the lowest three stages are located in the condenser neck. The lowest two extraction stage points are of the duplex type. The Bleed Steam System also includes vent paths from the MSRs, heater drain tank, and the FW heaters. For more details, refer to USAR (Section 10.2) and the May 22, 2003, application (Attachment 4, Section 8.3.2). Based on a review of the information that was submitted, no areas of the Bleed Steam System were identified as being impacted by the proposed stretch power uprate such that reactor safety considerations could be affected.

3.8.2.10 BOP - Other Systems, Flooding and Internally Generated Missiles

The following additional BOP-related systems were reviewed: Component Cooling Water System and Heater Drain System.

3.8.2.10.1 Component Cooling Water System

The Component Cooling Water (CCW) system is required to provide cooling water to various plant components during plant normal, shutdown, and post-accident operations. The CCW system also acts as an intermediate system between the components being cooled and the SW system.

As discussed in the May 22, 2003, application, Attachment 4, Section 8.3.11, the licensee has performed an evaluation of the CCW system and its components and has determined that the existing CCW system capability is adequate for the proposed stretch power uprate conditions with no equipment changes. The limiting heat loads for the CCW system occur during a normal plant cooldown, the 10 CFR 50 Appendix R plant cooldown, or during post-accident operations. The licensee indicated that the CCW system supply temperatures will not exceed the current operational limits as discussed in the May 22, 2003, application, Attachment 4 (Page 8-49) as long as the minimum required SW flow is maintained to the operating CCW system heat exchangers. The NRC staff has reviewed the licensee's evaluation of the CCW system, and based on the review, the NRC staff finds that the CCW system and its components will continue to perform their intended functions at the proposed 6.0-percent stretch power uprate conditions.

3.8.2.10.2 Heater Drain System

The Heater Drain System is a non-safety-related system that collects drains from the MS, Extraction Steam, and Bleed Steam systems. These drains are then returned to the Condensate System via either the condensers or the heater drain tank/pump. Based on a review of the information that was submitted, no areas of the Heater Drain System were identified as being impacted by the proposed stretch power uprate such that reactor safety considerations could be affected.

3.8.2.10.3 Flooding

The licensee has evaluated the impact of the proposed stretch power uprate on the existing flooding analysis and has determined that the flooding levels for the equipment at the current licensed power level remain unchanged. The licensee further concluded that any other equipment that was not previously flooded would not be adversely impacted by the stretch power uprate. Also, as discussed in the January 30, 2004, letter, Enclosure 2 (Pages 11 and 12), the licensee has assessed the impact of the proposed stretch power uprate on internal flooding and submergence levels due to postulated moderate and high energy line breaks (HELBs). The licensee has determined that flooding due to moderate energy line breaks remains bounded by the existing analysis since the operating conditions of moderate energy systems (pressures and temperatures) will not change during the stretch power uprate, and the volume of inventory sources that can cause flooding have not increased. Therefore, the licensee's internal flooding evaluation focused on the impact of postulated HELBs on the submergence levels identified in licensee's EQ plan. The FW line break outside containment is the bounding break for determining the maximum submergence levels. In the letter dated January 30, 2004, the licensee stated:

> The areas flooded and the associated submergence levels are based on specified break locations determined by the HELB criteria contained in USAR Appendix A and Fluor Power Services letter KPS-6601, 'Kewaunee Nuclear Power Plant Submergence Elevations- High-Energy Line Breaks.' The break locations for the FW lines are not impacted since the locations of peak pipe stress are not impacted and the physical line arrangements (terminal ends, branch connections, etc.) have not changed. These inputs are not impacted and are bounded by the existing evaluation. In addition, the maximum flood levels in all compartments, except the basement levels, are controlled by physical plant features such as doors and curb plates. Therefore, the submergence levels in these areas are not impacted by the power uprate.

Based on the licensee's assessment as discussed above, the NRC staff finds that the licensee has adequately considered and addressed any adverse impacts that the proposed stretch power uprate may have on the existing flooding analyses, and finds that the consequences of flooding and submergence levels remain the same and are not impacted by the proposed 6.0-percent stretch power uprate.

3.8.2.10.4 Internally Generated Missiles

In the letter dated January 30, 2004, Enclosure 2 (Pages 12 and 13), the licensee assessed the potential targets for internally generated missiles. The licensee indicated that the missile protection was designed to protect the safety-related systems, structures and equipment inside the containment from such missiles as might be generated in a LOCA for break sizes up to and including the double-ended severance of a main coolant pipe. The types of missiles for which protection is provided include: valve stems, valve bonnets, various types of bolts and nuts, pieces of pipes, etc. Removable slabs, blocks and partitions were evaluated to assure that the safety-related structures and equipment will not be affected during an operating basis earthquake or tornado. Further, the licensee addressed the impact of turbine missiles outside the containment, that could cause some localized damage. However, the licensee stated that since the turbine overspeed trip setpoint is being maintained below 120-percent (the current licensing basis), the effects of a turbine missile after the proposed stretch power uprate will not result in any significant damage.

Based on the analyses that were completed, the licensee has determined that the stretch power uprate will not adversely impact the evaluations that were performed for internally generated missiles or their conclusions. The proposed 6.0-percent stretch power uprate will not increase the postulated missiles that require consideration nor increase the equipment required to safely shutdown the plant following the generation of a missile. The NRC staff considers the licensee’s assessment to be acceptable for the proposed 6.0-percent stretch power uprate conditions.

3.8.2.11 Turbine-Generator

As discussed in Attachment 9 of the May 22, 2003, application and in the letter dated January 30, 2004, Siemens Westinghouse, the turbine-generator original equipment manufacturer (OEM), evaluated the performance of the turbine and generator at 1780 MWt (1772 MWt + 8 MWt RCP heat). The OEM’s evaluation of the main generator and generator support systems concluded that the generator and supporting systems are capable of supporting the proposed stretch power uprate provided the generator is operated within the existing generator capability curve and the SW temperature to the hydrogen coolers does not exceed 86 °F. The maximum design SW supply temperature is 80 °F. The OEM also found that if the generator hydrogen cooling valve is full open, the manual bypass valve may need to be throttled open to augment flow to the Hydrogen Coolers to maintain hydrogen gas temperatures within normal operating limits. The licensee’s evaluation of the turbine cited the OEM recommendations. Based on a review of the information that was submitted, the only area of the turbine-generator review that was impacted by the proposed stretch power uprate such that reactor safety considerations could be affected involves turbine overspeed protection.

As discussed in the May 22, 2003, application, Attachment 4 (Pages 8-12), the present turbine trip settings would allow the unit to overspeed to 121-percent of rated speed at the stretch power uprate conditions, which slightly exceeds the 120-percent turbine overspeed design requirement. Therefore, the licensee has made adjustments to the overspeed trip settings as recommended by Siemens-Westinghouse (the OEM). The OEM’s recommendations included: (1) reducing the maximum allowable settings for the mechanical overspeed trip from 111-percent to 109-percent, (2) reducing the maximum allowable settings for the electro-hydraulic

control (EHC) overspeed trip from 111.5-percent to 109.5-percent, and (3) reducing the maximum allowable settings for the redundant overspeed trip (ROST) from 111.5-percent to 109.5-percent, to ensure that the turbine-generator overspeed design criteria of 120-percent will not be exceeded at the proposed stretch power uprate conditions. As discussed in the letter dated January 30, 2004, Enclosure 2 (Page 14), the mechanical overspeed trip setting already satisfies the new setpoint requirement and did not require any adjustment, and the EHC and ROST setpoints were adjusted to satisfy the OEM's recommendations during the R26 refueling outage. Based on the licensee's assessment and implementation of the modifications that were recommended by the OEM as discussed above, the NRC staff finds that the licensee has adequately considered and addressed turbine-generator overspeed protection requirements for the proposed 6.0-percent stretch power uprate conditions and the NRC staff finds this acceptable.

3.8.3 Conclusion

The NRC staff has reviewed the licensee's safety analyses of the impact of the proposed stretch power uprate on (1) containment performance analyses and containment systems, (2) safe shutdown fire analyses and required systems, (3) spent fuel pool cooling analyses and systems, (4) flooding analyses, (5) NSSS interface systems, (6) ESF HVAC systems, and (7) safety-related cooling water systems. The NRC staff concludes that the results of licensee's analyses related to these areas would continue to meet the applicable acceptance criteria following implementation of the proposed stretch power uprate. Therefore, the NRC staff finds the proposed 6.0-percent stretch power uprate acceptable with respect to plant systems.

4.0 LICENSE AND TECHNICAL SPECIFICATION CHANGES

4.1 Change to Facility Operating License No. DPR-43

The licensee proposes to revise paragraph 2.C.(1) of the operating license, DPR-43, to authorize operation at reactor core power levels not in excess of 1772 MWt (100-percent power). The maximum core power level will be changed from 1673 MWt to 1772 MWt. This proposed change reflects the actual proposed power uprate in the core power level and is consistent with the results of the licensee's supporting safety analyses. Based on the evaluation provided in Section 3.0 above, the NRC staff finds the proposed change acceptable.

4.2 Change to TS 1.0m

The licensee proposes to revise the definition of "RATED POWER" in TS 1.0m to reflect the increase from 1673 MWt to 1772 MWt. The rated core power level will be changed from 1673 MWt to 1772 MWt. The TS change reflects the actual proposed change in the plant and it is consistent with the results of the licensee's supporting safety analyses. Based on the evaluation provided in Section 3.0 above, the NRC staff finds the proposed change acceptable.

4.3 Change to TS 2.1.c

The licensee proposes to revise TS 2.1.c, regarding peak fuel centerline temperature, to increase the peak fuel centerline temperature from < 4700 °F to < 5080 °F. Additionally, the licensee proposes the following text to be inserted at the end of TS 2.1.c, “decreasing by 58 °F per 10,000 MWD/MTU of burnup.”

The basis is the change to Westinghouse fuel. The licensee proposes to change TS 2.1.c to be consistent with NUREG-1431, “Standard Technical Specifications Westinghouse Plants,” Volume 1, Revision 1, April 1995.

10 CFR Part 50, Appendix A, GDC 10 requires that SAFDLs are not exceeded during normal operation and AOOs. This is accomplished by having a DNB design basis which corresponds to a 95 percent probability at a 95 percent confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below melting temperature. The reactor core safety limits are established to prevent violation of these criteria.

The licensee proposes to revise TS 2.1.c regarding peak fuel centerline temperature. The proposed change revises the peak fuel centerline temperature limit from < 4700 °F to < 5080 °F (and decreasing by 58 °F per 10,000 MWD/MTU of burnup). The fuel centerline temperature limit of 5080 °F decreasing by 58 °F per 10,000 MWD/MTU of burnup has been the fuel melting limit for Westinghouse fuel since the mid-1960's. This limit is an accepted limit for Westinghouse fuel and is supported by experimental data and Westinghouse technical evaluations. The Westinghouse data is generically applicable to all uranium dioxide fuel, and as such, is applicable to the co-resident Framatome ANP fuel remaining in the KNPP core during the transition cycles to an all Westinghouse fuel core. The NRC staff approved the fuel centerline temperature limit as part of WCAP-12610-P-A (WCAP-12610-P-A, “VANTAGE+ Fuel Assembly Reference Core Report,” dated April 1995) and WCAP-14483-A (WCAP-14483-A, “Generic Methodology for Expanded Core Operating Limits Report,” dated January 1999). Based on this, the NRC staff finds that the proposed fuel centerline temperature limit TS change is acceptable for KNPP.

Based on the evaluation provided in Section 3.0 above, the NRC staff finds the proposed changes acceptable.

4.4 Change to Table TS 3.5-1

The licensee proposes to revise Table 3.5-1, “Engineered Safety Features Initiation Instrument Setting Limits,” setting limit for 6, “High-High Steam Flow in a Steam Line Coincident with Safety Injection,” from 4.5 X 10^6 pounds per hour (lb/hr) at 735 pounds per square inch gauge (psig) to 4.4 X 10^6 lb/hr at 735 psig.

The basis of the change to the high-high steam flow TS setting is the change in the analytical limit used in the high-high steam flow setpoint calculation. This calculation documented the total loop accuracy for the high-high steam flow actuation logic circuit and was used to determine the TS limit and the actual plant setting for the high-high steam flow trip. The analytical limit in the setpoint uncertainty calculation corresponds with the maximum calibrated span of the plant’s steam flow transmitters. The actual plant setting for high-high steam flow trip is within the loop calibrated span which ensures the trip will occur prior to reaching the

analytical limit. The accident analyses assume a higher high-high steam flow trip (7.76 X 10^6 lb/hr), which causes the trip to actuate later in the accident analysis and is conservative.

Based on the evaluation provided in Section 3.0 above, the NRC staff finds the proposed changes acceptable.

4.5 Change to TS 3.3.c.1.A.3 (iii)

The licensee proposes to delete in its entirety TS 3.3.c.1.A.3 (iii), which allowed both containment fancoil unit trains to be out of service for 72 hours provided both containment spray trains remain operable, and the subsequent item, (iv), renumbered as (iii).

Based on the evaluation provided in Section 3.0 above, the NRC staff finds the proposed changes acceptable.

4.6 Change to TS 3.4.c.1 and TS 3.4.c.2

The licensee proposes to revise TS 3.4.c.1, Condensate Storage Tank, to increase the minimum volume from 39,000 gallons to 41,500 gallons. The licensee also proposes to add “Usable volume” to TS 3.4.c.1 for clarification. The licensee proposed to revise TS 3.4.c.2, Condensate Storage Tank, to increase the minimum volume from 39,000 gallons to 41,500 gallons. The licensee also proposes to add “Usable volume” to TS 3.4.c.2 for clarification.

The basis of these changes is the SBO 4-hour coping period. The stretch power uprate evaluations determined that the increased decay heat resulting from the increased core power required more condensate storage tank inventory during the 4-hour coping period.

Both of the TSs for the Condensate Storage Tank, have been revised to increase the minimum volume from 39,000 gallons to 41,500 gallons. “Usable volume” is added to the specification for clarification. According the licensee, the basis for these changes is the SBO 4-hour coping period. The decay heat increase as a result of the proposed stretch power uprate requires more CST inventory during the 4-hour coping period. The evaluations associated with these changes demonstrate continued safe operation of the plant. Therefore, the NRC staff finds the proposed changes acceptable.

4.7 Change to TS 3.4.b

The licensee proposes to revise TS 3.4.b regarding the AFW system to require three operable AFW trains prior to increasing reactor power above 1673 MWt. AFW trains are defined in the TS Bases Section 3.4.b, page TS B3.4-1. The licensee also proposes to revise the following: (1) TS 3.4.b.2 would be renumbered as 3.4.b.4, (2) New TS 3.4.b.3 would be inserted stating the following: “The reactor power shall not be increased above 1673 MWt unless three trains of AFW are OPERABLE. If two of three AFW trains are inoperable, then within 2 hours, action shall be taken to reduce reactor power to $\leq$ 1673 MWt.”, (3) TS 3.4.b.3 would be renumbered as 3.4.b.5 and the reference to TS 3.4.b.2 in the last sentence would be revised to reference TS 3.4.b.3 and TS 3.4.b.4, (4) TS 3.4.b.4 would be renumbered as 3.4.b.2, (5) TS 3.4.b.5 would be renumbered as 3.4.b.6 and reference to TS 3.4.b.2 would be replaced with TS 3.4.b.4, and (6) TS 3.4.b.6 would be renumbered as 3.4.b.7.

As discussed in Section 3.8.2.5 of this SE, the licensee has determined that for the most limiting LONF event, one AFW pump may not be able to supply enough FW to dissipate the increased reactor decay heat load that results from operation at the proposed uprated power conditions. For all other accident and transient scenarios, the licensee has determined that one AFW pump will continue to be sufficient for the uprated power condition. Currently, TS 3.4.b.2 states:

> " When the Reactor Coolant System temperature is > 350 °F, any of the following conditions of inoperability may exist during the time interval specified:
>
> A. One auxiliary feedwater train may be inoperable for 72 hours.
> B. Two auxiliary feedwater trains may be inoperable for 4 hours.
> C. One steam supply to the turbine-driven auxiliary feedwater pump may be inoperable for 7 days."

Thus, the current TS requirement would allow one or more AFW pumps to be inoperable (as provided above) while reactor power is being increased above the current licensed reactor power limit. In order to assure that the AFW system will be able to dissipate reactor decay heat for the worst-case LONF event during operation at the uprated reactor power level, the licensee proposes to revise TS 3.4.b to require three AFW trains to be operable prior to increasing reactor power above the current licensed reactor power limit of 1673 MWt. In order to accomplish this, the licensee proposed a new paragraph into TS 3.4.b, which states:

> "The reactor power shall not be increased above 1673 MWt unless three trains of AFW are OPERABLE. If two of three AFW trains are inoperable, then within two hours, reduce reactor power to ≤ 1673 MWt."

Other than the above, the licensee did not modify any of the current TS requirements, except for re-ordering the requirements contained within the subsections of TS 3.4.b. The proposed changes to TS 3.4.b and associated Bases are depicted on Pages TS 3.4-2, and TS B 3.4-2 through TS B 3.4-4, respectively (May 22, 2003, application, Attachment 2).

In order to determine the adequacy of the AFW system for the proposed stretch power uprate conditions, the licensee completed two LONF event analyses: one at the current power limit of 1673 MWt., and another at the proposed stretch power uprate limit of 1772 MWt. The results of the licensee's analyses demonstrated that: (a) one AFW pump provides sufficient heat removal capacity to mitigate the LONF accident at 1673 MWt.; and (b) two of the three AFW pumps are required to provide adequate AFW flow capacity to dissipate the decay heat that could result from the worst-case LONF accident while operating at 1772 MWt. Consequently, three AFW trains must be operable during operation at power levels above 1673 MWt. in order to mitigate the worst-case LONF accident, assuming a concurrent limiting single active failure, consistent with the plant licensing basis criteria.

Another observation from the licensee's analysis is that the turbine-driven AFW pump may not have sufficient capacity to mitigate the worst-case LONF event during the proposed stretch power uprate conditions. Consequently, the AFW system as is, does not fully satisfy the post-TMI Action Plan criteria that would have two full capacity AFW trains with diverse power sources. However, as discussed in the letter dated January 30, 2004, Enclosure 2 (Page 18), the licensee indicated that the RCPs contribute 8 MWt to the RCS when they are operating. If

the RCPs are tripped by the reactor operator following a LONF event, the capacity of the turbine-driven AFW pump will be sufficient to mitigate the worst-case LONF event. Based on discussions with the licensee, the KNPP Emergency Operating Procedures (in conjunction with Emergency Contingency Procedures) will assure appropriate operator action to trip the RCPs following a LONF event if necessary for assuring adequate decay heat removal. Also, for LONF events that involve a loss-of-offsite power, the RCPs would trip and a single AFW pump would be capable of mitigating the event.

Based on the above considerations, the NRC staff agrees that the proposed change to TS 3.4.b is necessary and adequate to assure continued compliance with the existing KNPP licensing basis requirements during reactor operation at the proposed uprated power level. The new 2-hour time period that is proposed for reducing power to below the uprated power condition when two AFW pumps are inoperable is appropriate, because the licensee's analysis indicates that (without relying on operator action to trip the RCPs) the AFW system may not be able to fully mitigate the worst-case LONF event, and two hours provides sufficient time for the operators to reduce reactor power to below the uprate power level. The NRC staff also finds that the provisions contained in the KNPP Emergency Operating Procedures (as referred to above and discussed with the licensee) are adequate to assure that the turbine-driven AFW pump will be able to mitigate the worst-case LONF event without a concurrent loss-of-offsite power if neither of the two motor-driven AFW pumps are operable. Consequently, the NRC staff finds that the proposed changes to TS 3.4.b are acceptable for the proposed 6.0-percent stretch power uprate.

During review of the proposed change to TS 3.4.b, the NRC staff questioned the licensee's implementation of the existing requirements of TS 3.4.b.2 (listed above). While the TS Bases made it clear that the steam supplies from both SGs had to be operable as a condition of operability of the turbine-driven AFW pump, the NRC staff noted that TS 3.4.b.2 was not stated very clearly such that the TS requirements could easily be misinterpreted and applied in a non-conservative manner. Consistent with the Standard Technical Specification requirements, the NRC staff's interpretation of the KNPP TS is that if one steam supply to the turbine-driven AFW pump becomes inoperable, the turbine-driven AFW pump is considered to be inoperable and the TS allows a 7-day allowed outage time to resolve the problem (see Item C, above). However, if a motor-driven AFW pump becomes inoperable during the period when the steam supply to the turbine-driven AFW pump is inoperable, two AFW pumps are considered to be inoperable. That is to say, in addition to being in the 7-day action requirement (Condition C) for the inoperable steam supply to the turbine-driven AFW pump and entering the 72-hour action requirement for the inoperable motor-driven AFW pump (see Condition A, above), the 4-hour action requirement for two inoperable AFW pumps would also apply (see Condition B, above). The licensee indicated that TS 3.4.b.2 was not being implemented in this manner at KNPP and that for the condition described, only Conditions A and C would be entered. This is of concern to the NRC staff because by applying the licensee's interpretation, the plant could be in a 72-hour action requirement with only one motor-driven AFW pump operable, and not be able to mitigate the design-basis Main Steam Line Break event inside containment. This would be the case any time that the operable motor-driven AFW pump is aligned to the same SG as the operable steam supply to the turbine-driven AFW pump, and the fault is associated with that same SG. The licensee acknowledged the NRC staff's concern and documented it in their Corrective Action Process (CAP) for tracking and resolution (CAP020151, dated February 25, 2004). Because this concern is generally applicable irrespective of the licensee's power uprate request, the NRC staff considers the licensee's actions to document the concern in the KNPP

CAP program for tracking and resolution to be appropriate and acceptable. The NRC staff has also informed the NRC Resident Inspector for KNPP of the concern for appropriate NRC follow-up action.

Based on this review, the NRC staff concludes that there are no significant hazards associated with the proposed TS changes; therefore, the NRC staff finds the TS changes for the proposed stretch power uprate acceptable.

4.8 Change to TS 2.3.a.3.A, TS 2.3.a.3.B, and Table TS 4.1-2

The licensee proposed to revise TS 2.3.a.3.A for f(ΔI), to change “An even function” to “A function.” The licensee also proposes to revise TS 2.3.a.3.A, for f(ΔI), to change f(ΔI) to f_1(ΔI). The licensee also proposes to revise TS 2.3.a.3.B, for f(ΔI), to change f(ΔI) to f_2(ΔI). In addition, the licensee proposes to revise Table TS 4.1-2, “Minimum Frequencies for Sampling Tests,” to change the units in the frequency column for sampling test 7, “Secondary Coolant, b. Iodine Concentration,” from 0.1 µCi/cc to 0.1 µCi/gram.

The regulation at 10 CFR Part 50, Appendix A, GDC-10 requires that SAFDLs are not exceeded during steady state operation, normal operational transients, and AOOs. This is accomplished by having a DNB design-basis (95/95 DNB criterion) that DNB will not occur on the limiting fuel rods, and by requiring that fuel centerline temperature stays below the melting temperature. The reactor core safety limits are established to preclude violation of these criteria. Automatic enforcement of the reactor core safety limits is provided by the RPS, which includes a number of reactor trip functions, two of which are the OTΔT and OPΔT reactor trips. The design of the OTΔT reactor trip function provides protection against violating the TS safety limit for DNB ratio (DNBR). The OPΔT reactor trip function ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1 percent cladding strain) under all possible overpower conditions.

KNPP TS 2.3.a.3.A and TS 2.3.a.3.B define the Overtemperature ΔT and Overpower ΔT setpoint equations, including the constant and parameters values. All constant and parameter values for these setpoint equations are specified in the KNPP COLR. KNPP employs NRC approved methodology outlined in WCAP-8745-P-A (WCAP-8745-P-A, “Design Basis for the Thermal Overtemperature ΔT and Thermal Overpower ΔT Trip Functions,” dated September 1986) to calculate these parameter values. This NRC staff-approved methodology is currently referenced in KNPP TS 6.9. In accordance with NRC GL 88-16 (NRC GL 88-16, “Removal of Cycle-Specific Parameter Limits from Technical Specifications (GL 88-16),” dated October 3, 1988), the licensee can revise these values using a NRC staff approved methodology, without prior NRC review and approval. In accordance with this approved methodology, the licensee is revising the full power ΔT_o inputs to both the Overtemperature ΔT and Overpower ΔT setpoints, and is revising the K1 constant and f(ΔI) function inputs in the Overtemperature ΔT setpoint. The revised values are KNPP Cycle 26 Reload SE values for the stretch power uprate conditions. The licensee verified that the Overtemperature ΔT and Overpower ΔT setpoint values assumed in the USAR Chapter 14 transient analyses which support the stretch power uprate bound these setpoint input changes.

The licensee is proposing editorial changes to TS 2.3.a.3.A and TS 2.3.a.3.B regarding the f(ΔI) function. TS 2.3.a.3.A is being revised for f(ΔI) to change “An even function” to “A function.”

This change is incorporated for accuracy, as the f(ΔI) function is not an even function because f(-ΔI) does not equal f(+ΔI). Therefore the word "even" is being deleted. Additionally, TS 2.3.a.3.A is being revised to change f(ΔI) to f_1(ΔI), and TS 2.3.a.3.B is being revised to change f(ΔI) to f_2(ΔI). These changes are simply editorial in nature and are consistent with the Westinghouse Standard Technical Specifications (NUREG-1431, Revision 2, "Standard Technical Specifications Westinghouse Plants," dated June 2001. The NRC staff finds these TS changes to be acceptable.

5.0 REGULATORY COMMITMENTS

To support the proposed KNPP 6.0-percent stretch power uprate, the licensee made the following commitments, which are to be included in the KNPP Operating License as license conditions. The licensee has committed to:

1) Revise all documents, as appropriate, for the stretch power uprate to address Framatome ANP fuel DNBR design basis prior to implementation of the license amendment.

2) Complete changes to the condensate storage tank level setpoints, and first-stage turbine pressure, as appropriate, prior to implementation of the license amendment.

3) Complete piping and pipe support evaluations for service water and component cooling water prior to implementation of the license amendment.

4) Update the KNPP EQ Plan, as appropriate, to reflect power uprate evaluations prior to implementation of the license amendment.

5) Revise plant procedures, as appropriate, to accommodate the stretch power uprate prior to implementation of the license amendment. Emergency, abnormal, and operating procedures that are entered due to a loss-of-normal feedwater event or have AFW TS requirements shall be changed, as appropriate, prior to implementation of the license amendment.

6) Review new TS requirements, revised procedures, and any control room changes due to the stretch power uprate to determine necessary changes to the operator training program and complete the required portion of the operator training prior to implementation of the license amendment.

7) Update setpoint changes for reactor protection and control inputs, alarms, computer constants, and embedded values, consistent with operation at 1772 MWt prior to implementation of the license amendment. Power range nuclear instruments shall be recalibrated and checked based on a secondary heat balance prior to implementation of the license amendment.

8) Revise degraded voltage and thrust calculations for MOV operators outside containment which were reviewed for impact of uprated post accident temperatures, as required, prior to implementation of the license amendment.

9) Provide the NRC with a status update and schedule for resolution of GL 96-06 water hammer issues by April 2, 2004.

10) Incorporate the increase in flow rate and velocities, as well as the changes in operating pressures and temperatures, into the KNPP FAC Program as part of the power uprate implementation. The KNPP FAC Program models shall be updated prior to the next program inspections scheduled for the next refueling outage.

11) Establish an inspection and monitoring program to monitor potential feedwater heater degradation at the stretch power uprate conditions. An inspection program shall be developed based on the baseline inspection results and using programs and processes in place at KNPP. This shall be completed prior to the next refueling outage.

The NRC staff considered the above commitments as part of its evaluation in Section 3.0 above and finds the commitments appropriate for the proposed 6.0-percent stretch power uprate. The NRC staff has conditioned the proposed 6.0-percent stretch power uprate on completion of the above commitments.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Wisconsin State official was notified of the proposed issuance of the amendment. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (68 FR 34670). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

8.0 CONCLUSION

The NRC staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Attachments: 1) List of Acronyms
2) Tables 1 through 10

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Date: February 27, 2004

LIST OF ACRONYMS

AAC	alternate alternating current
AC	alternating current
ACRS	Advisory Committee on Reactor Safeguards
AEC	Atomic Energy Commission
AFW	auxiliary feedwater
AMSAC	anticipated transient without scram mitigating system actuation circuitry
ANSI	American National Standards Institute
AGO	anticipated operational occurrences
AOP	Abnormal Operating Procedures
ART	adjusted reference temperature
ARV	atmospheric relief valves
ASME	American Society of Mechanical Engineers
AST	alternate source term
ASTM	American Society for Testing and Materials
ATC	American Transmission Company
ATWS	anticipated transient without scram
BOP	balance-of-plant
CCW	component cooling water
CDF	core damage frequency
CFCU	containment fancoil unit
CFR	*Code of Federal Regulations*
CIA	containment integrity analysis
COLR	core operating limits report
CRDM	control rod drive mechanism
CRDS	control rod drive system
CST	condensate storage tank

CUF	cumulative usage factor
CVCS	chemical and volume control system
CW	circulating water
DBA	design-basis accident
DC	direct current
DEHL	double ended hot-leg
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
EAB	exclusion area boundary
DSS	diverse scram system
ECCS	emergency core cooling system
EDG	emergency diesel generator
EFPY	effective full-power year
EOP	emergency operating procedure
EQ	equipment qualification
ESF	engineered safety feature
ESFAS	engineered safety feature actuation system
FAC	flow-accelerated corrosion
FHA	fuel-handling accident
FRV	feedwater regulating valves
FW	feedwater
FWP	feedwater pumps
GDC	general design criteria
GDT	gas decay tank
GL	generic letter
GOTHIC	Generation of Thermal-Hydraulic Information for Containment
HELB	high-energy line break
HP	high pressure

HVAC	heating, ventilation, and air conditioning
I&C	instrumentation and control
ISA	independent safety analysis
IST	inservice testing
KNPP	Kewaunee Nuclear Power Plant
LBB	leak before break
LBLOCA	large-break loss-of-coolant accident
LHSI	low-head safety injection
LOCA	loss-of-coolant accident
LONF	loss of normal feedwater
LP	low pressure
LPZ	low population zone
LPI	low-pressure injection
LSSS	limiting safety system setting
LTC	long-term cooling
LTOP	low-temperature overpressure protection
MAT	main auxiliary transformer
MDLM	mist diffusion layer model
MFIV	main feedwater isolation valves
MSLB	main steamline break
MOV	motor-operated valve
MS	main steam
MSIV	main steam isolation valve
MSSV	main steam safety valves
MT	main transformer
MTC	moderator temperature coefficient
MUR	measurement uncertainty recapture
MVA	megavolts-amperes

MWe	megawatts electric
MWt	megawatts thermal
NC	natural circulation
NPSH	net positive suction head
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
OEM	original equipment manufacturer
PLP	possible loose parts
PORV	power-operated relief valve
PT	pressure-temperature
PTS	pressurized thermal shock
PPC	plant process computer
PPCS	plant process computer screen
PWR	pressurized-water reactor
PWSCC	primary water stress corrosion cracking
RAI	request for additional information
RAT	reserve auxiliary transformer
RCCA	rod cluster control assembly
RCL	reactor coolant loop
RCP	reactor coolant pressure
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RG	regulatory guide
RHR	residual heat removal
RPS	reactor protection system
RSG	replacement steam generator
RSICC	Radiation Safety Information and Computation Center
RTD	resistance temperature detector

RTDP	revised thermal design procedure
RTO	reactor thermal output
RTP	rated thermal power
RWST	refueling water storage tank
SAFDL	specified acceptable fuel design limit
SAL	safety analysis limit
SBO	station blackout
SBLOCA	small-break loss-of-coolant accident
SDS	steam dump system
SE	safety evaluation
SER	safety evaluation report
SFPCS	spent fuel pool cooling system
SFP	spent fuel pool
SG	steam generator
SGBD	steam generator blowdown system
SGTR	steam generator tube rupture
SI	safety injection
SIS	safety injection system
SRP	Standard Review Plan
SRSS	square root of the sum of the squares
SSC	structure, system, and component
SWS	service water system
TS	technical specification
TSC	technical support center
TT	thermally treated
UFMD	ultrasonic flow measurement device
UPI	upper plenum injection
USAR	updated safety analysis report

USE	upper shelf energy
UTM	ultrasonic temperature measurement
VCT	volume control tank
VAC	volts alternating current

Table 1
Design-Basis Accident Licensee Calculated Radiological Consequences
TEDE[15] (rem)

Design-Basis Accident	EAB[16]	LPZ[17]	Control Room
MSLB, Pre-existing iodine spike	0.03	0.01	0.50
Dose acceptance criteria[18]	25	25	5
MSLB, Accident-initiated iodine spike	0.06	0.02	1.00
Dose acceptance criteria	2.5	2.5	5
Locked Rotor Accident	0.50	0.08	1.40
Dose acceptance criteria	2.5	2.5	5
Control Rod Ejection Accident	0.40	0.09	1.91
Dose acceptance criteria	6.3	6.3	5
SGTR, Pre-existing spiking	1.30	0.30	3.10
Dose acceptance criteria	25	25	5
SGTR, Accident-initiated spiking	0.80	0.20	1.00
Dose acceptance criteria	2.5	2.5	5
LBLOCA, total	1.31	0.22	4.58
Dose acceptance criteria	25	25	5
FHA	0.70	0.11	1.00
Dose acceptance criteria	6.3	6.3	5

[15] Total Effective Dose Equivalent
[16] Exclusion Area Boundary, licensee reported as site boundary dose
[17] Low Population Zone
[18] Dose acceptance criteria taken from SRP 15.0.1 and GDC-19

Table 2
Assumptions Used in Radiological Consequence Analysis
Main Steamline Break

Reactor coolant activity	
Pre-existing iodine spike case	60.0 µCi/gm DE I-131[1]
Accident-initiated iodine spike case	1.0 µCi/gm DE I-131
Accident-initiated iodine appearance rate spiking factor	500 times equilibrium rate
Duration of accident-initiated iodine spike	4 hours
Secondary coolant activity	0.1 µCi/gm DE I-131
Primary coolant mass	1.19E+08[2] gm
Secondary coolant initial liquid mass	
Faulted steam generator (SG)	161,000 lbm
Intact SG	84,000 lbm
Steam release from faulted SG	161,000 lbm
Time to release faulted SG initial mass	2 minutes
Steam release from intact SG	
0 - 2 hours	222,000 lbm
2 - 8 hours	424,000 lbm
8 - 24 hours	614,000 lbm
Time to cool RCS and stop faulted SG release	72 hours
Steam partition coefficient	
Faulted steam generator	1
Intact steam generator	0.01
Steam generator tube leak rate	150 gallons per day per SG
Time until begin control room emergency HVAC	5 minutes
Normal ventilation flow rates	
Unfiltered makeup	2500 scfm (±10%)
Emergency ventilation system flow rates	
Filtered makeup	0 scfm
Filtered recirculation	2500 scfm (±10%)
Unfiltered inleakage	200 scfm
Control room filter efficiencies	
Elemental	90%
Organic	90%
Particulate	99%
Atmospheric dispersion factors	Table 10

[1]Dose Equivalent Iodine-131

[2]1.19E+08 = 1.19 x 10^8 = 119,000,000

Table 3
Assumptions Used in Radiological Consequence Analysis
Locked RCP Rotor

Reactor coolant activity	60.0 µCi/gm DE I-131
Secondary coolant activity	0.1 µCi/gm DE I-131
Primary coolant mass	1.19E+08 gm
Secondary coolant mass	7.89E+07 gm
Fuel rods in core failing	100%
No fuel melting	
Fission product gap fractions	
I-131	0.08
Kr-85	0.10
Other iodines and noble gases	0.05
Alkali metals	0.12
Iodine chemical form in release	97% elemental, 3% organic
Primary-to-secondary SG tube leak rate	150 gallons per day per SG
Steam release from secondary	
0 - 2 hours	210,000 lbm
2 - 8 hours	455,000 lbm
Steam partition coefficient	0.01
Time to cool RCS and stop steam release	8 hours
Time until begin control room emergency HVAC	10 minutes
Normal ventilation flow rates	
Unfiltered makeup	2500 scfm (±10%)
Emergency ventilation system flow rates	
Filtered makeup	0 scfm
Filtered recirculation	2500 scfm (±10%)
Unfiltered inleakage	200 scfm
Control room filter efficiencies	
Elemental	90%
Organic	90%
Particulate	99%
Atmospheric dispersion factors	Table 10

Table 4
Assumptions Used in Radiological Consequence Analysis
Control Rod Ejection Accident

Reactor power	1782.6 MWt
Reactor coolant activity	60.0 µCi/gm DE I-131
Secondary coolant activity	0.1 µCi/gm DE I-131
Primary coolant mass	1.19E+08 gm
Secondary coolant mass	7.89E+07 gm
Radial peaking factor	1.7
Fuel rods in core failing	15%
Fission product gap fractions	
Iodines and noble gases	0.10
Alkali metals	0.12
Fuel rods in core melting	0.375%
Fission product activity released from melted fuel	
Noble gases and alkali metals	100%
Iodines	25% for containment leakage path 50% for SG steaming path
SG steaming release pathway	
Primary-to-secondary SG tube leak rate	150 gallons per day per SG
Steam release from secondary	
0 - 200 seconds	800 lbm/sec
200 - 1800 seconds	100 lbm/sec
> 1800 seconds	0 lbm/sec
Steam partition coefficient	0.01
Iodine chemical form in steam release	97% elemental, 3% organic
Containment leakage pathway	
Containment net free volume	1.32E+06 ft^3
Shield building volume	3.74E+05 ft^3
Shield building participation fraction	0.5
Containment leak rate	
0 - 24 hours	0.5 weight %/day
> 24 hours	0.25 weight %/day
Containment leak path fractions	
0 -10 minutes	
Through shield building	0.0
Through auxiliary building	0.1
Direct to environment	0.9
> 10 minutes	
Through shield building	0.89
Through auxiliary building	0.1
Direct to environment	0.01

Table 4 (continued)
Assumptions Used in Radiological Consequence Analysis
Control Rod Ejection Accident

Shield building air flow	
0 - 10 minutes	
Shield building to environment	Not applicable
Shield building recirculation	Not applicable
10 - 30 minutes	
Shield building to environment	6000 scfm (±10%)
Shield building recirculation	0.0 scfm
> 30 minutes	
Shield building to environment	3100 scfm
Shield building recirculation	2300scfm
Shield building and auxiliary building filter efficiencies	
Elemental	90%
Organic	90%
Particulate	99%
Time until begin control room emergency HVAC	2.5 minutes
Normal ventilation flow rates	
Unfiltered makeup	2500 scfm (±10%)
Emergency ventilation system flow rates	
Filtered makeup	0 scfm
Filtered recirculation	2500 scfm (±10%)
Unfiltered inleakage	200 scfm
Control room filter efficiencies	
Elemental	90%
Organic	90%
Particulate	99%
Atmospheric dispersion factors	Table 10

Table 5
Assumptions Used in Radiological Consequence Analysis
Steam Generator Tube Rupture

Reactor coolant activity	
Pre-existing iodine spike case	60.0 µCi/gm DE I-131
Accident-initiated iodine spike case	1.0 µCi/gm DE I-131
Accident-initiated iodine appearance rate spiking factor	500 times equilibrium rate
Duration of accident-initiated iodine spike	4 hours
Secondary coolant activity	0.1 µCi/gm DE I-131
Primary coolant mass	1.19E+08 gm
Secondary coolant initial liquid mass	84,000 lbm/SG
Intact steam generator tube leak rate	150 gallons per day
Pre-trip releases (< 173.3 seconds)	
Tube rupture break flow	16,900 lbm
Percentage of break flow that flashes to steam	19.93%
Steam release to condenser	1077.8 lbm/sec for each SG
Post-trip releases (> 173.3 seconds)	
Tube rupture break flow	138,000 lbm
Percentage of break flow that flashes to steam	14.76%
Steam release from ruptured SG, 0 - 2 hours	86,400 lbm
Steam release from intact SG, 0 - 2 hours	233,400 lbm
Steam release from intact SG, 2 - 8 hours	488,800 lbm
Steam release from intact SG, 8 - 24 hours	662,800 lbm
Steam partition coefficient	
Ruptured steam generator, break flow	1
Intact steam generator	0.01
Time until begin control room emergency HVAC	5 minutes
Normal ventilation flow rates	
Unfiltered makeup	2500 scfm (±10%)
Emergency ventilation system flow rates	
Filtered makeup	0 scfm
Filtered recirculation	2500 scfm (±10%)
Unfiltered inleakage	200 scfm
Control room filter efficiencies	
Elemental	90%
Organic	90%
Particulate	99%
Atmospheric dispersion factors	Table 10

Table 6
Assumptions Used in Radiological Consequence Analysis
Large-Break Loss-of-Coolant Accident

Reactor power	1782.6 MWt
Source term	Based on RG 1.183
Containment volume	1.32E+06 ft^3
Containment leak rate	
0 - 24 hours	0.5 weight % per day
> 24 hours	0.25 weight % per day
Shield building volume	3.74E+05 ft^3
Shield building participation fraction	0.5
Containment leak modeling	See Table 4
Spray operation	
Time to initiate sprays	0.0 hours
Termination of sprays	0.91 hours
Recirculation spray	Not credited
Removal coefficients	
Elemental iodine	20 hr^{-1}
Particulate	4.5 hr^{-1}
Sedimentation (after spray termination)	0.1 hr^{-1}
ECCS leakage	
Containment sump volume	315,000 gal
ECCS leak rate, 0 - 30 days	12 gal/hr
Airborne percent iodine to auxiliary building	
0 - 3 hours	10%
> 3 hours	1%
ECCS leak rate to RWST	
0 - 24 hours	3 gpm
> 24 hours	1.5 gpm
Shield and auxiliary building filter efficiencies	
Elemental	90%
Organic	90%
Particulate	99%
Time until begin control room emergency HVAC	2 minutes
Normal ventilation flow rates	
Unfiltered makeup	2500 scfm ($\pm$10%)
Emergency ventilation system flow rates	
Filtered makeup	0 scfm
Filtered recirculation	2500 scfm ($\pm$10%)
Unfiltered inleakage	200 scfm
Control room filter efficiencies	
Elemental	90%
Organic	90%
Particulate	99%
Atmospheric dispersion factors	Table 10

Table 7
Assumptions Used in Radiological Consequence Analysis
Fuel Handling Accident

Reactor power	1782.6 MWt
Radial peaking factor	1.7
Fission product decay period	100 hours
Number of fuel assemblies damaged	1
Fuel pool water depth	23 ft
Pool iodine effective decontamination factor	200
Fuel gap fission product inventory	
I-131	8%
Kr-85	10%
Other iodines and noble gases	5%
Duration of release	2 hours
Time until begin control room emergency HVAC	1 minute
Normal ventilation flow rates	
Unfiltered makeup	2500 scfm (±10%)
Emergency ventilation system flow rates	
Filtered makeup	0 scfm
Filtered recirculation	2500 scfm (±10%)
Unfiltered inleakage	200 scfm
Control room filter efficiencies	
Elemental	90%
Organic	90%
Particulate	99%
Atmospheric dispersion factors	Table 10

Table 8
Waste Gas Decay Tank Rupture

Assumptions

Noble gas inventory	See submittals licensing report Table 6.7-14.
Release duration	5 minutes
Time until begin control room emergency HVAC	30 seconds
Normal ventilation flow rates	
Unfiltered makeup	2500 scfm (±10%)
Emergency ventilation system flow rates	
Filtered makeup	0 scfm
Filtered recirculation	2500 scfm (±10%)
Unfiltered inleakage	0 scfm
Control room filter efficiencies	
Elemental	90%
Organic	90%
Particulate	99%
Atmospheric dispersion factors	Table 10

Licensee's Calculated Results

Location	TEDE (rem)
EAB	0.10
LPZ	0.02
Control Room	0.80

Table 9
Volume Control Tank Rupture

Assumptions

Noble gas inventory	See submittals licensing report Table 6.7-15.
Release duration	5 minutes
Primary coolant noble gas activity	1% fuel defect level
Primary coolant initial iodine activity	60 µCi/gm DE I-131
Letdown flow rate	88 gpm
Partitioning of iodine for spilled water	10%
Time until begin control room emergency HVAC	30 seconds
Normal ventilation flow rates	
Unfiltered makeup	2500 scfm (±10%)
Emergency ventilation system flow rates	
Filtered makeup	0 scfm
Filtered recirculation	2500 scfm (±10%)
Unfiltered inleakage	0 scfm
Control room filter efficiencies	
Elemental	90%
Organic	90%
Particulate	99%
Atmospheric dispersion factors	Table 10

Licensee's Calculated Results

Location	TEDE (rem)
EAB	0.10
LPZ	0.01
Control Room	0.40

Table 10
Atmospheric Dispersion Factors

Exclusion Area Boundary

Time (hr)	X/Q (sec/m^3)
0 - 2	2.232E-04

Low Population Zone

Time (hr)	X/Q (sec/m^3)
0 - 2	3.977E-05
2 - 24	4.100E-06
24 - 48	2.427E-06
48 - 720	4.473E-07

Control Room

Time (hr)	X/Q (sec/m^3)
0 - 8	2.93E-03
8 - 24	1.73E-03
24 - 48	6.74E-04
48 - 720	1.93E-04

March 8, 2007

Mr. David A. Christian
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Chief Nuclear Officer
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Glen Allen, VA 23060-6711

SUBJECT: KEWAUNEE POWER STATION - ISSUANCE OF AMENDMENT RE: RADIOLOGICAL ACCIDENT ANALYSIS AND ASSOCIATED TECHNICAL SPECIFICATIONS CHANGE (TAC NO. MC9715)

Dear Mr. Christian:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 190 to Facility Operating License No. DPR-43 for the Kewaunee Power Station. This amendment revises the Technical Specifications in response to your application dated January 30, 2006, as supplemented by letter dated January 23, 2007.

The amendment revises radialogical accident analyses and associated technical specifications.

A copy of the NRC staff's Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice.

Sincerely,

/RA/

Robert F. Kuntz, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-305

Enclosures:
1. Amendment No. 190 to
   License No. DPR-43
2. Safety Evaluation

cc w/encls: See next page

March 8, 2007

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Senior Vice President and
Chief Nuclear Officer
Innsbrook Technical Center
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*See SE **NLO with comments

OFFICE	NRR/LPL3-1/PM	NRR/LPL3-1/LA	NRR/AADB/BC	NRR/SCVB/BC	OGC	NRR/LPL3-1/BC
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DATE	3/07/07	3/06/07	2 /22/07	7/21/07	2/28/07	3/08/07

OFFICIAL RECORD COPY

Kewaunee Power Station

cc:

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DOMINION ENERGY KEWAUNEE, INC.

DOCKET NO. 50-305

KEWAUNEE POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 190
License No. DPR-43

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Dominion Energy Kewaunee, Inc. dated January 30, 2006 as supplemented by letter dated January 23, 2007, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-43 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 190, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Lakshminaras Raghavan, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility Operating License
and Technical Specifications

Date of Issuance: March 8, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 190

FACILITY OPERATING LICENSE NO. DPR-43

DOCKET NO. 50-305

Replace the following page of the Facility Operating License No. DPR-43 with the attached revised page. The changed area is identified by a marginal line.

REMOVE	INSERT
Page 3	Page 3

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
TS 3.1-7	TS 3.1-7
TS 3.6-4	TS 3.6-4
TS 6.20-1	TS 6.20-1

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR, Chapter 1: (1) Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70, (2) is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and (3) is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady-state reactor core power levels not in excess of 1772 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 190, are hereby incorporated in the license. The licensee shall operate the facility In accordance with the Technical Specifications.

(3) Fire Protection

The licensee shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the licensee's Fire Plan, and as referenced in the Updated Safety Analysis Report, and as approved in the Safety Evaluation Reports, dated November 25, 1977, and December 12, 1978 (and supplement dated February 13, 1981) subject to the following provision:

The licensee may make changes to the approved Fire Protection Program without prior approval of the Commission, only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(4) Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Nuclear Management Company Kewaunee Nuclear Power Plant Physical Security Plan (Revision 0)" submitted by letter dated October 18, as supplemented by letter dated October 21, 2004, July 26, 2005, and May 15, 2006.

(5) Fuel Burn-up

The maximum rod average Burn-up for any rod shall be limited to 60 GWD/MTU until completion of an NRC environmental assessment supporting an increased limit.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 190 TO FACILITY OPERATING LICENSE NO. DPR-43

DOMINION ENERGY KEWAUNEE, INC.

KEWAUNEE POWER STATION

DOCKET NO. 50-305

1.0 INTRODUCTION

By letter dated January 30, 2006 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML060540217), as supplemented by letter dated January 23, 2007 (ADAMS Accession No. ML070240543) Dominion Energy Kewaunee (DEK, the licensee) requested an amendment to facility Operating License No. DPR-43 for the Kewaunee Power Station (KPS). The licensee proposed to modify the currently approved radiological accident analyses and associated technical specifications (TSs). This proposed amendment incorporates TS changes to compensate for the higher control room emergency zone (CREZ) unfiltered inleakage measured during the American Society for Testing and Materials (ASTM) E741 (tracer gas) leakage test conducted in December 2004.

Results from the ASTM testing of the KPS control room envelope showed the CREZ unfiltered inleakage to be greater than that assumed in the approved radiological accident analyses. The revised CREZ unfiltered inleakage was determined to be a facility change, which caused an increase in the dose consequences of the approved radiological accident analyses. Currently, the KPS CREZ is operable but non-conforming. The resolution of this condition is to incorporate the increase in assumed CREZ unfiltered inleakage into the radiological accident analyses.

The supplemental letter contained clarifying information, did not change the initial no significant hazards consideration determination, and did not expand the scope of the original *Federal Register* notice.

The licensee proposed changes to the TSs for certain plant parameters to compensate for the higher measured CREZ unfiltered inleakage. This safety evaluation (SE) addresses the Nuclear Regulatory Commission (NRC) staff's review of the licensee's revised radiological accident analyses. In this license amendment request, the licensee proposed to change:

1. TS 3.1.c.2.A, "Maximum Coolant Activity," coolant activity limit that requires intermediate shutdown from 60 μCi/gram DOSE EQUIVALENT I-131 to 20 μCi/gram DOSE EQUIVALENT I-131.

2. TS 3.6.c.3.B, "Performance Requirement," Shield Building Ventilation System and the Auxiliary Building Special Ventilation system filter removal efficiency from $\geq$95 percent radioactive methyl iodide removal to $\geq$97.5 percent radioactive methyl iodide removal.

3. TS 6.20, "Containment Leakage Rate Testing Program," maximum allowable leakage rate from 0.5 weight percent of the contained air per 24 hours at the peak test pressure (P_a) of 46 psig to 0.2 weight percent.

2.0 REGULATORY EVALUATION

The U.S. Nuclear Regulatory Commission (NRC) staff finds that the licensee in Section 5.2 of its January 30, 2006, submittal, identified the applicable regulatory requirements. The regulatory requirements and guidance which the staff considered in its review of the requested action are as follows:

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criterion 19 (GDC-19) requires, in part, that "[a] control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident."

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36 "Technical specifications," requires that "an applicant for a license authorizing operation of a production or utilization facility include proposed technical specifications" in its license application.

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.67 "Accident source term" establishes analyzed dose limits for acceptable adoption of the accident source term. Title 10 of the *Code of Federal Regulations* (10 CFR) 50.67(b)(2) states that "[t]he NRC may issue the amendment only if the applicant's analysis demonstrates with reasonable assurance that:

(i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem)[22] total effective dose equivalent (TEDE).

(ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) TEDE.

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[2]The use of 0.25 SV (25rem) TEDE is not intended to imply that this value constitutes an acceptable limit for emergency doses to the public under accident conditions. Rather, this 0.25 SV (25 rem) TEDE value has been stated in this section as a reference value, which can be used in the evaluation of proposed design basis changes with respect to potential reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation.

(iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident."

The NRC staff also considered the guidance contained in Regulatory Guide (RG) 1.183, "Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," RG 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors" as well as NUREG-0800 "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP) Section 15.0.1"Radiological Consequence Analyses Using Alternative Source Terms" in reviewing the licensee's amendment request.

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the licensee's analysis methods, assumptions, and inputs using docketed information provided by the licensee.

These radiological accident analyses of record for the KPS license were previously docketed in Amendment No. 166, issued March 17, 2003 (ADAMS Accession No. ML030210062), which implemented an alternate source term; and Amendment No. 172, issued February 27, 2004 (ADAMS Accession No. ML040430633), which implemented a stretch power uprate to 1772 mega-watt thermal (MWt). These previously approved radiological accident analyses used the analytical methods and assumptions outlined in RG 1.183.

The revised radiological accident analyses for design-basis accidents (DBA) incorporate changes for the control room isolation parameters based on air flow measurements of ASTM E741 (tracer gas) testing conducted in response to Generic Letter 2003-01.

The control room envelope unfiltered inleakage was measured by tracer gas testing on December 14 and 15, 2004. The leak test measured 409 ± 29 cubic feet per minute (cfm) for system train A, and 447 ± 51 cfm for train B. In the revised radiological accident analyses, DEK adjusted the control room unfiltered inleakage rate during normal mode heating ventilation and air conditioning (HVAC) operation to range between 1620 to 2750 cubic feet per minute (cfm). DEK specified two unfiltered inleakage for control room emergency ventilation. For events that model isolation actuated by a safety injection (SI) set point, CREZ unfiltered inleakage was assumed to be at least 800 cfm. For events actuated by high radiation monitor in the control room air supply duct, the CREZ unfiltered inleakage was assumed to be at least 1500 cfm. Isolation actuated by the control room air supply duct radiation monitor does not close all control room isolation dampers. The unfiltered inleakage rate of 1500 cfm compensates for the dampers that remain open. DEK also increased the assumed control room damper closure time to 20 seconds from the previous value of 10 seconds. The increased damper closure time bounds actual measured closure times.

DEK compensated for the higher control room unfiltered inleakage in the radiological accident analysis by proposing modifications to TSs limits. The three proposed TS changes reduce the calculated fission product released to the environment, thus allowing for higher control room unfiltered inleakage. In this amendment request, DEK submitted revised radiological analyses of the DBAs. The dose acceptance criteria for the DBAs and the revised licensee-calculated radiological consequence are listed in Table 1. The CREZ unfiltered inleakage parameters and assumptions used by the licensee and acceptable to the NRC staff are listed in Table 2.

3.1 Main Steamline Break (MSLB) Accident

In the revised radiological analysis for MSLB accident, DEK changed the assumed CREZ unfiltered inleakage from 200 to 1000 cfm, which bounds the ASTM E741 test results. This is the only assumption that has changed from the previous radiological accident analysis of MSLB.

The licensee assumed that the faulted steam generator (SG) boils dry within 2 minutes. The entire liquid inventory of the faulted SG is steamed off and all the iodine initially in the SG is released to the outside environment. The primary-to-secondary SG tube leakage rate is assumed to be at the TS limit of 150 gallons per day (gpd) per SG. The 150 gpd leakage for the faulted SG, along with its noble gas and iodine, is assumed released directly to the outside atmosphere. In the intact SG, the 150 gpd primary-to-secondary leakage mixes with the bulk SG secondary coolant water. Transferred noble gases are released without holdup, and iodine is released to the outside environment at the steaming rate of the intact SG, with credit for partitioning when the SG tubes are covered with water.

DEK analyzed the MSLB for two iodine spiking cases. The pre-accident iodine spiking case assumed that a reactor transient has occurred prior to the MSLB, and has raised the reactor coolant system (RCS) iodine concentration to 60 micro curies per gram (µCi/gm) of dose equivalent (DE) Iodine 131 (I-131). The accident-initiated iodine spiking case assumed that the reactor trip associated with the MSLB creates an increase in the iodine release rate from the fuel to the coolant to a value 500 times greater than the release rate corresponding to a maximum equilibrium RCS concentration of 1.0 µCi/gm DE I-131. The accident-initiated spike duration is 4 hours. The secondary coolant activity in both cases is assumed to be the TS limit of 0.1 µCi/gm DE I-131. No fuel damage is projected for the MSLB.

The low steamline pressure SI set point will be reached shortly after the onset of an MSLB. The SI signal causes the control room HVAC to switch from normal operation mode to the accident mode of operation. DEK conservatively assumed that the control room HVAC does not fully enter the accident mode of operation until 5 minutes after the event begins.

The NRC staff reviewed the licensee's analysis of the MSLB radiological consequences, and finds that they remain consistent with the guidance provided in RG 1.183. The licensee's calculated radiological consequences at the Exclusion Area Boundary (EAB), Low Population Zone (LPZ) and in the KPS control room are within the dose criteria specified in 10 CFR 50.67 and GDC-19, and are within the acceptance criteria given in SRP 15.0.1 for the MSLB. The NRC staff finds the results of the licensee's calculations (Table 1), and the major parameters and assumptions used by the licensee (Table 3) acceptable.

3.2 Locked Rotor Accident

In the revised locked rotor accident analysis, DEK changed the assumption for the fraction of failed fuel rods from 100 percent down to 50 percent, which is less conservative. DEK based the 50 percent assumption on the reload safety analysis limit. DEK also increased the length of time assumed for control room HVAC to enter accident mode of operation from 10 to 45 minutes. DEK also conservatively revised the CREZ unfiltered inleakage to 1500 cfm based on tracer gas test results, and because the control room air supply duct radiation monitor actuates the HVAC accident mode of operation. These are the only assumptions that have changed

from the previous radiological accident analysis of the locked rotor accident.

The licensee's analysis assumes that a reactor transient has occurred prior to the locked rotor accident and that the transient has raised the RCS iodine activity concentration to 60 μCi/gm DE I-131, which bounds the proposed TS 3.1.c.2.A limit of 20 μCi/gm DE I-131. The noble gas and alkali metal activity concentration in the primary coolant is based on a fuel defect level of 1.0 percent. The iodine activity concentration in the secondary coolant is assumed to be 0.1 μCi/gm DE I-131, and the alkali metal activity concentration is assumed to be 10 percent of the primary coolant concentration. Accident-induced activity is assumed to be released to the environment as a result of primary-to-secondary leakage through the SG tubes and steaming from the secondary side, released through either the atmospheric relief valves or safety valves. An iodine partitioning factor in the SGs of 0.01 is used to account for retention of iodine in the SG as the water turns to steam. The partitioning factor of 0.01 is also applied to the alkali metal activity release. All noble gas activity carried over to the secondary side of the SGs is assumed to be immediately released to the outside atmosphere. At 8 hours after the accident, the licensee assumed that the residual heat removal (RHR) system has removed all decay heat with no further releases to the environment after that time.

The NRC staff reviewed the licensee's methods, inputs and assumptions used in its revised radiological consequences analysis of the locked rotor accident and finds that they are consistent with the guidance given in RG 1.183. The licensee's calculated radiological consequences at the EAB, LPZ and in the KPS control room are within the dose limits specified in 10 CFR 50.67 and GDC-19, and are within the acceptance criteria given in SRP 15.0.1 for the locked rotor accident. The NRC staff finds the results of the licensee's calculations (Table 1), and the major parameters and assumptions used by the licensee (Table 4) acceptable.

3.3 Control Rod Ejection Accident

In the revised radiological analysis of the control rod ejection accident, DEK maintained the same assumptions as applied in previous radiological analysis, except the CREZ unfiltered inleakage rate has been conservatively increased to 1000 cfm to account for tracer gas test results.

This DBA postulates the mechanical failure of a control rod drive mechanism pressure housing that results in the ejection of a rod cluster control assemble and drive shaft. Localized damage to fuel cladding and a limited amount of fuel melting are projected. The radioactivity in the primary coolant is assumed to leak through the SG tubes into the secondary coolant. A portion of this activity is released to the outside atmosphere through the main condenser, atmospheric relief valves or safety valves. Additionally, radioactive primary coolant is discharged to the containment through the opening in the reactor vessel head where the control rod assembly was ejected. The activity in the containment is assumed to be released to the environment as a result of design-basis containment leakage evaluated at the proposed TS limit of 0.5 percent per day for the first 24 hours. After that, the containment is assumed to leak at half that rate until the end of the 30-day period considered in the analysis. In each case, the containment

and secondary coolant release pathways are considered separately with bounding source term release for the combined release path ways.

DEK assumed that 15 percent of the fuel rods in the core suffer sufficient damage such that all their gap activity is released. The licensee assumed that 10 percent of the total core activity of iodine and noble gases and 12 percent of the total core activity for alkali metals are in the fuel gap, consistent with guidance provided in RG 1.183. A small fraction of the fuel in the failed rods is assumed to melt as a result of the rod ejection. The licensee estimated this melting to be limited to 0.375 percent of the core. This estimate was previously accepted in amendment No.166 (ML030210062).

The licensee assumed 100 percent of noble gases and alkali metals in the failed fuel gap and melted fuel are released to either the RCS or the containment, depending on the pathway assumed. For the containment leakage pathway, the licensee assumed that all the iodine from the gap of the failed fuel and 25 percent of the iodine released from melted fuel are released to the containment atmosphere. For the primary-to-secondary leakage release pathway, the licensee assumed that all the iodine from the gap of the failed fuel and 50 percent of the iodine released from melted fuel is released to the RCS.

As discussed for the locked rotor accident (Section 3.2 of this SE), the licensee assumed an iodine partitioning factor of 0.01 in the SGs for the primary-to-secondary leakage release pathway. For the containment leakage release pathway, no credit was taken for iodine or particulate removal mechanisms.

The low pressurizer pressure SI setpoint is expected to be reached within 60 seconds of the onset of the control rod ejection. The SI signal causes the control room HVAC to switch from the normal operation mode to the accident mode of operation. For this accident, DEK conservatively assumed that the control room HVAC does not fully enter the accident mode of operation until 2.5 minutes after the event begins.

The NRC staff reviewed the licensee's methods, inputs and assumptions used in its revised radiological consequences analysis of the control rod ejection accident and finds that they remain consistent with the conservative guidance given in RG 1.183. The licensee's calculated radiological consequences at the EAB, LPZ and in the KPS control room are within the dose limits specified in 10 CFR 50.67 and GDC-19 and are within the acceptance criteria given in SRP 15.0.1 for the control rod ejection accident. The NRC staff finds the results of the licensee's calculations (Table 1), and the major parameters and assumptions used by the licensee (Table 5) acceptable. To verify the licensee's dose results the NRC staff performed confirmatory calculations for the control rod ejection accident and finds the licensee's results to be reasonable.

3.4 Steam Generator Tube Rupture (SGTR)

In the revised radiological analysis of the SGTR, DEK changed the assumed pre-accident iodine spike from a value of 60 μCi/gm DE I-131 to a value of 20 μCi/gm DE I-131 per TS 3.1.c.2.A. DEK also increased the CREZ unfiltered inleakage from 200 to 1000 cfm based on the tracer gas test results.

The SGTR is analyzed for two iodine spiking cases; a pre-existing iodine spike resulting in

elevated primary coolant activity, and an iodine spike initiated by the accident. For the pre-existing iodine spike case, the RCS iodine activity concentration is assumed to be at the proposed TS 3.1.c.2.A limit for a transient, equal to 20 μCi/gm DE I-131. For the accident initiated iodine spike case, the associated reactor trip causes an increase in the iodine release rate from the fuel to the RCS to a value 500 times the rate associated with the TS equilibrium RCS activity concentration of 1.0 μCi/gm DE I-131. The duration of the accident initiated iodine spike is limited by the amount of iodine in the fuel gap. Based on having 8 percent of the core inventory of iodine in the fuel gap, the spike would last 4 hours. RG 1.183 allows an accident initiated spiking factor of 335 for the SGTR, and the NRC staff finds the licensee's assumed factor of 500 is conservative compared to the RG value. All other analysis inputs are consistent with the guidance in RG 1.183.

The low pressurizer pressure SI setpoint is expected to be reached at around 2.9 minutes after the onset of the SGTR. The SI signal causes the control room HVAC to switch from the normal operation mode to the accident mode of operation. The licensee conservatively assumed that the control room HVAC does not fully enter the accident mode of operation until 5 minutes after the event begins.

The NRC staff reviewed the licensee's methods, inputs and assumptions used in its revised radiological consequences analysis of the SGTR and finds that they are consistent with the conservative guidance given in RG 1.183. The licensee's calculated radiological consequences at the EAB, LPZ and in the KPS control room, are within the dose limits specified in 10 CFR 50.67 and GDC-19 and are within the acceptance criteria given in SRP 15.0.1 for the SGTR. The NRC staff finds the results of the licensee's calculations (Table 1), and the major parameters and assumptions used by the licensee (Table 6) acceptable.

3.5 Large-Break Loss-of-Coolant Accident (LBLOCA)

For the LBLOCA analysis, the current radioactive methyl iodide removal percentage TS limit for the shield building ventilation system, and the Auxiliary Building Special Ventilation System carbon filters is ≥ 95 percent. The licensee reviewed historical data of radiological accident analysis (RAA) sensitivity cases, to ensure that although more limiting, the proposed conservative change to ≥ 97.5 percent is reasonable and continues to provide adequate operating margin. The NRC staff finds that the revised limits bound plant charcoal filter test results and provide sufficient operating margin and are therefore acceptable.

For the LBLOCA analysis, the containment leakage rate is reduced to 0.2 weight percent per day of the contained air per 24 hours at a peak test pressure of 46 psig from the current analysis value of 0.5 weight percent per day. The licensee reviewed historical data of RAA sensitivity cases, to ensure that although more limiting, the proposed conservative change is reasonable and continues to provide adequate margin to actual measurements of containment leakage rates. The NRC staff finds that the revised containment leak rate limit bounds the plant measured containment leak rate test result, and provides sufficient operating margin and is therefore acceptable.

In its revised radiological analysis of LBLOCA, DEK changed the assumed shield building and auxiliary building filter efficiencies, the containment leakage rate, and the CREZ unfiltered inleakage flow rate. The revised assumptions increased shield building and auxiliary building

filter efficiencies from 90 percent to 95 percent for removal of both elemental and organic iodine. The containment leak rates are revised from 0.5 to 0.2 weight percent per day for the first 24 hours, per proposed TS 6.20, and from 0.25 to 0.1 weight percent per day for greater than 24 hours. The CREZ unfiltered inleakage flow rate is revised from 200 to 800 cfm based on tracer gas test results. These are the only assumptions that have changed from the previous radiological accident analysis of LBLOCA.

In the licensee's analysis of the LBLOCA radiological consequences, activity from the damaged core is released into the containment. Three pathways for release to the environment are considered in the analysis:

(1) design-basis containment leakage,
(2) leakage from engineering safety feature (ESF) systems outside containment, and
(3) emergency core cooling system (ECCS) recirculation back-leakage to the refueling water storage tank (RWST).

The calculated radiological consequences of these three release pathways are added together to determine the total LBLOCA radiological consequences.

3.5.1 Containment Leakage Pathways

The containment is assumed to leak at the proposed TS 6.20 design-basis leak rate of 0.2 percent per day for the first 24 hours of the accident, and then to leak at half that rate for the remainder of the 30-day analysis period. The licensee assumed that during the first 10 minutes of the accident, 90 percent of the activity leaking from the containment is discharged directly to the environment. The remaining 10 percent enters the auxiliary building where it is released through filters. After 10 minutes, only 1 percent of the activity leaking from the containment is assumed to go directly to the environment, 10 percent continues to go to the auxiliary building, and 89 percent is assumed to go into the shield building. The air discharged from the shield building is filtered. Additionally, once the shield building is brought to sub-atmospheric pressure at 30 minutes into the accident, iodine and particulate can be removed by recirculation through filters. A shield building participation fraction of 0.5 is assumed.

The shield building filter efficiency for elemental and organic iodine is revised to 95 percent, which is bound by the proposed level in TS 3.6.c.3.B. The licensee assumed removal of iodine through sedimentation for particulate and the containment spray for elemental and particulate forms of iodine. The KPS containment spray system is an ESF system and is designed to provide containment cooling and fission product removal in the containment following a LBLOCA. One train of spray was assumed to operate. Switch over to recirculation spray is not credited and all spray removal is terminated when the RWST drains down at 0.91 hours from the start of the accident. In determining the core spray iodine removal rates, the licensee assumed a reduction in assumed spray flow relative to that assumed in the analysis supporting Amendment No. 166 (ML030210062). This reduction is intended to bound potential pump degradation. The NRC staff finds this change acceptable. The licensee assumed a sedimentation coefficient of 0.1 hr^{-1} for particulate after the core spray system is terminated.

The licensee used the models and guidance provided in RG 1.183 and SRP 6.5.2, “Containment Spray as a Fission Product Cleanup System,” to determine the removal rates for iodine.

3.5.2 Post-LOCA ESF Leakage Pathway

During the recirculation phase of long-term core cooling, radioactive water from the containment sump is sent to ECCS equipment located outside the containment. These components may leak into the auxiliary building. Although ECCS recirculation does not occur until 0.91 hours after the accident begins, the licensee conservatively assumed leakage occurs immediately upon the onset of the LBLOCA. The licensee conservatively assumed the leakage to the auxiliary building is 12 gallons per hour. The licensee assumed that 10 percent of the activity in the leaked fluid becomes airborne when the sump temperature is above 212 °F. Once the sump temperature drops below 212 °F at 3 hours from the start of the event, the airborne activity fraction is reduced to 1 percent of the activity in the leaked fluid. The assumed time that the sump temperature falls below 212 °F is selected to bound the results of the containment response analyses performed for the TS changes. The NRC staff finds these assumptions are consistent with RG 1.183 and are acceptable. The licensee also assumed that half of the airborne iodine activity in the auxiliary building is removed by plateout on surfaces. This assumption was previously approved in Amendment No. 166 (ML030210062).

3.5.3 ECCS Back-Leakage to the RWST

RHR back-leakage to the RWST is assumed to be at a rate of 3 gallons per minute (gpm) for the first 24 hours, and 1.5 gpm for the remainder of the accident. It is assumed that 1 percent of the iodine becomes airborne, even when the sump temperature is above 212 °F since any incoming water would be cooled by the water remaining in the RWST.

3.5.4 Control Room Ventilation System Modeling

For the LBLOCA, the low pressurizer pressure SI setpoint will be reached shortly after the start of the event. The SI signal causes the control room HVAC to switch from the normal operation mode to the accident mode of operation. The licensee conservatively assumed that the control room HVAC does not fully enter the accident mode of operation until 2 minutes after the event begins.

3.5.5 LBLOCA Conclusion

The NRC staff reviewed the licensee’s methods, inputs and assumptions used in its revised radiological consequences analysis of the LBLOCA for TS changes, and finds that they are consistent with the conservative guidance given in RG 1.183. The licensee’s calculated radiological consequences at the EAB, LPZ and in the KPS control room are within the dose limits specified in 10 CFR 50.67 and GDC-19, and are within the acceptance criteria given in SRP 15.0.1 for the LBLOCA. The NRC staff finds the results of the licensee’s calculations (Table 1), and the major parameters and assumptions used by the licensee (Table 7) acceptable. To verify the licensee’s dose results, the NRC staff performed confirmatory radiological consequence analyses of the LBLOCA and finds the licensee’s results to be reasonable.

3.6 Fuel-Handling Accident (FHA)

In the revised radiological analysis, DEK conservatively changed the CREZ unfiltered inleakage flow rate from 200 to 1500 cfm. This is the only assumption that has changed from the previous radiological accident analysis of FHA.

The licensee's analysis of the FHA was performed with assumptions selected so that the results are bounding for an accident that occurs either in the containment or in the auxiliary building. Activity released from the damaged assembly is assumed to be released to the environment through either the containment purge system or the spent fuel pool ventilation system, without credit for filtration or isolation of the containment, containment purge system, or spent fuel pool ventilation system. The decay time used, 100 hours, is the minimum decay time required by TS before movement of fuel. The licensee assumed that all the fuel rods in the equivalent of one fuel assembly are damaged, and all the gap activity in the rods is released to the pool. A pool iodine effective decontamination factor of 200 is assumed. All fuel gap noble gas activity is assumed to be released from the pool. All activity released from the pool is assumed to be released to the outside environment within 2 hours.

DEK assumed that the control room HVAC system is in normal operation mode at the onset of the FHA. A high-radiation signal for the control room air supply duct is generated as a result of the activity release to the atmosphere, and control room HVAC enters accident mode of operation within 25 minutes.

The NRC staff reviewed the licensee's methods, inputs and assumptions, and finds that they are consistent with the conservative guidance given in RG 1.183. The licensee's calculated radiological consequences at the EAB, LPZ and in the KPS control room, are within the dose limits specified in 10 CFR 50.67 and GDC-19 and are within the acceptance criteria given in SRP 15.0.1 for the FHA. The NRC staff finds the results of the licensee's calculations (Table 1), and the major parameters and assumptions used by the licensee (Table 8) acceptable. To verify the licensee's dose results, the NRC staff performed a confirmatory radiological consequence analysis of the FHA and finds the results to be reasonable.

3.7 Waste Gas Decay Tank (GDT) Rupture and Volume Control Tank (VCT) Rupture

The KPS licensing basis includes analyses of the radiological consequences of the waste GDT rupture, and the VCT rupture. The radiological analyses for these two accidents were previously found acceptable by the NRC staff in its SE approving Amendment No. 166 to the KPS license. The only change to the assumptions in the existing analyses is the increase in CREZ unfiltered inleakage. The NRC staff documented in its SE approving Amendment No. 172, issued February 27, 2004 (ADAMS Accession No. ML040430633) that "[b]ecause of the short duration of the radiation release, minimizing the assumed control room unfiltered inleakage maximizes the calculated control room dose. This is due to less dilution of the radioactivity in the control room. The NRC staff finds these assumptions to be acceptable based on plant operation and the operation of the radiation monitoring and control room HVAC systems." Therefore, the assumed increase in the unfiltered inleakage for these two accidents is bounded by the previous analyses. Consequently, these two accidents were not re-analyzed.

3.8 Technical Evaluation Conclusion

The NRC staff reviewed the assumptions, inputs, and methods used by DEK to reevaluate the radiological consequences of DBAs. The NRC staff finds that DEK used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2. DEK's analysis demonstrated that the radiological consequences of DBAs would remain within applicable regulatory limits. The NRC staff has performed confirmatory calculations on selected accidents and finds, with reasonable assurance, that the revised radiological accident analyses, assuming higher CREZ unfiltered inleakage and amended TS, (TS 3.1.c.2.A, TS3.6.c.3.B, and TS 6.20) complies with the regulatory requirements and is therefore acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Wisconsin State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

This 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 staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding (71 FR 13172). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

6.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: D. Chung, B. Lee

Date: March 8, 2007

Table 1
Design-Basis Accident Licensee Calculated Radiological Consequences
TEDE (rem)

Design-Basis Accident	EAB	LPZ	Control Room
MSLB, Pre-existing iodine spike	0.03	0.01	0.70
Dose acceptance criteria	25	25	5
MSLB, Accident-initiated iodine spike	0.06	0.02	2.60
Dose acceptance criteria	2.5	2.5	5
Locked Rotor Accident	0.40	0.06	3.90
Dose acceptance criteria	2.5	2.5	5
Control Rod Ejection Accident	0.40	0.09	4.54
Dose acceptance criteria	6.3	6.3	5
SGTR, Pre-existing spiking	0.50	0.10	1.90
Dose acceptance criteria	25	25	5
SGTR, Accident-initiated spiking	0.80	0.20	2.80
Dose acceptance criteria	2.5	2.5	5
LBLOCA, total	0.52	0.09	4.95
Dose acceptance criteria	25	25	5
FHA	0.90	0.15	4.0
Dose acceptance criteria	6.3	6.3	5

Table 2
Revised Control Room Parameters

	REVISED ASSUMPTION	PREVIOUS ASSUMPTION
Normal ventilation flow rates		
Unfiltered Makeup Flow Rate	1620 - 2750 cfm	2250 - 2750 cfm
Emergency Ventilation Flow Rates		
Unfiltered Inleakage Following SI	$\geq$ 800 cfm	200 cfm
Unfiltered Inleakage Following R-23[3]	1500 cfm	200 cfm
Control Room Isolation Damper Closure Time	20 seconds	10 seconds

[3] The CREZ UFI is increased to at least 1500 cfm for events that model control room isolation on a control room radiation monitor, R-23, high control room duct activity monitor actuation (i.e. locked rotor and fuel handling accident).

Table 3
Assumptions Used in Radiological Consequence Analysis
Main Steamline Break

Reactor coolant activity	
Pre-existing iodine spike case	60.0 μCi/gm DE I-131
Accident-initiated iodine spike case	1.0 μCi/gm DE I-131
Accident-initiated iodine appearance rate spiking factor	500 times equilibrium rate
Duration of accident-initiated iodine spike	4 hours
Secondary coolant activity	0.1 μCi/gm DE I-131
Primary coolant mass	1.19E+08 gm
Secondary coolant initial liquid mass	
Faulted steam generator (SG)	161,000 lbm
Intact SG	84,000 lbm
Steam release from faulted SG	161,000 lbm
Time to release faulted SG initial mass	2 minutes
Steam release from intact SG	
0 - 2 hours	222,000 lbm
2 - 8 hours	424,000 lbm
8 - 24 hours	614,000 lbm
Time to cool RCS and stop faulted SG release	72 hours
Steam partition coefficient	
Faulted steam generator	1
Intact steam generator	0.01
Steam generator tube leak rate	150 gallons per day per SG
Time until begin control room emergency HVAC	5 minutes
Normal ventilation flow rates	
Unfiltered makeup	2500 cfm (+10%)
Emergency ventilation system flow rates	
Filtered makeup	0 cfm
Filtered recirculation	2500 cfm (+10%)
Unfiltered inleakage	1000 cfm
Control room filter efficiencies	
Elemental	90%
Organic	90%
Particulate	99%
Atmospheric dispersion factors	Table 9

Table 4
Assumptions Used in Radiological Consequence Analysis
Locked Rotor Accident

Reactor coolant activity	60.0 µCi/gm DE I-131
Secondary coolant activity	0.1 µCi/gm DE I-131
Primary coolant mass	1.19E+08 gm
Secondary coolant mass 0 to 30 minutes Secondary coolant mass > 30 minutes	7.89E+07 gm 1.06E+08 gm
Secondary coolant mass > 30 minutes	1.06E+8 gm
Fuel rods in core failing, No fuel melting	50%
Peaking Factor Applied to Calculate Activity in Failed Fuel Rods	1.7
Fission product gap fractions	
I-131	0.08
Kr-85	0.10
Other iodines and noble gases	0.05
Alkali metals	0.12
Iodine chemical form in release	97% elemental, 3% organic
Primary-to-secondary SG tube leak rate	150 gallons per day per SG
Steam release from secondary	
0 - 2 hours	210,000 lbm
2 - 8 hours	455,000 lbm
Steam partition coefficient	0.01
Time to cool RCS and stop steam release	8 hours
Time until begin control room emergency HVAC	45 minutes
Normal ventilation flow rates	
Unfiltered makeup	2500 cfm (+10%)
Emergency ventilation system flow rates	
Filtered makeup	0 cfm
Filtered recirculation	2500 cfm (+10%)
Unfiltered inleakage	1500 cfm
Control room filter efficiencies	
Elemental	90%
Organic	90%
Particulate	99%
Atmospheric dispersion factors	Table 9

Table 5
Assumptions Used in Radiological Consequence Analysis
Control Rod Ejection Accident

Reactor power	1782.6 MWt
Reactor coolant activity	60.0 μCi/gm DE I-131
Secondary coolant activity	0.1 μCi/gm DE I-131
Primary coolant mass	1.19E+08 gm
Secondary coolant mass	7.89E+07 gm
Radial peaking factor	1.7
Fuel rods in core failing	15%
Fission product gap fractions	
Iodines and noble gases	0.10
Alkali metals	0.12
Fuel rods in core melting	0.375%
Fission product activity released from melted fuel	
Noble gases and alkali metals	100%
Iodines	25% for containment leakage path 50% for SG steaming path
SG steaming release pathway	
Primary-to-secondary SG tube leak rate	150 gallons per day per SG
Steam release from secondary	
0 - 200 seconds	800 lbm/sec
200 - 1800 seconds	100 lbm/sec
> 1800 seconds	0 lbm/sec
Steam partition coefficient	0.01
Iodine chemical form in steam release	97% elemental, 3% organic
Containment leakage pathway	
Containment net free volume	1.32E+06 ft^3
Shield building volume	3.74E+05 ft^3
Shield building participation fraction	0.5
Containment leak rate	
0 - 24 hours	0.5 weight %/day
> 24 hours	0.25 weight %/day

Table 5 (continued)
Assumptions Used in Radiological Consequence Analysis
Control Rod Ejection Accident

Containment leak path fractions	
0 -10 minutes	
Through shield building	0.0
Through auxiliary building	0.1
Direct to environment	0.9
> 10 minutes	
Through shield building	0.89
Through auxiliary building	0.1
Direct to environment	0.01
Shield building air flow	
0 - 10 minutes	
Shield building to environment	Not applicable
Shield building recirculation	Not applicable
10 - 30 minutes	
Shield building to environment	6000 cfm (+10%)
Shield building recirculation	0.0 cfm
> 30 minutes	
Shield building to environment	3100 cfm
Shield building recirculation	2300cfm
Shield building and auxiliary building filter efficiencies	
Elemental	90%
Organic	90%
Particulate	99%
Time until begin control room emergency HVAC	2.5 minutes
Normal ventilation flow rates	
Unfiltered makeup	2500 cfm (+10%)
Emergency ventilation system flow rates	
Filtered makeup	0 cfm
Filtered recirculation	2500 cfm (+10%)
Unfiltered inleakage	1000 cfm
Control room filter efficiencies	
Elemental	90%
Organic	90%
Particulate	99%
Atmospheric dispersion factors	Table 9

Table 6
Assumptions Used in Radiological Consequence Analysis
Steam Generator Tube Rupture

Reactor coolant activity	
Pre-existing iodine spike case	20.0 μCi/gm DE I-131
Accident-initiated iodine spike case	1.0 μCi/gm DE I-131
Accident-initiated iodine appearance rate spiking factor	500 times equilibrium rate
Duration of accident-initiated iodine spike	4 hours
Secondary coolant activity	0.1 μCi/gm DE I-131
Primary coolant mass	1.19E+08 gm
Secondary coolant initial liquid mass	84,000 lbm/SG
Intact steam generator tube leak rate	150 gallons per day
Pre-trip releases (< 173.3 seconds)	
Tube rupture break flow	16,900 lbm
Percentage of break flow that flashes to steam	19.93%
Steam release to condenser	1077.8 lbm/sec for each SG
Post-trip releases (> 173.3 seconds)	
Tube rupture break flow	138,000 lbm
Percentage of break flow that flashes to steam	14.76%
Steam release from ruptured SG, 0 - 2 hours	86,400 lbm
Steam release from intact SG, 0 - 2 hours	233,400 lbm
Steam release from intact SG, 2 - 8 hours	488,800 lbm
Steam release from intact SG, 8 - 24 hours	662,800 lbm
Steam partition coefficient	
Ruptured steam generator, break flow	1
Intact steam generator	0.01
Time until begin control room emergency HVAC	5 minutes
Normal ventilation flow rates	
Unfiltered makeup	2500 cfm (+10%)
Emergency ventilation system flow rates	
Filtered makeup	0 cfm
Filtered recirculation	2500 cfm (+10%)
Unfiltered inleakage	1000 cfm
Control room filter efficiencies	
Elemental	90%
Organic	90%
Particulate	99%
Atmospheric dispersion factors	Table 9

Table 7
Assumptions Used in Radiological Consequence Analysis
Large-Break Loss-of-Coolant Accident

Reactor power	1782.6 MWt
Source term	Based on RG 1.183
Containment volume	1.32E+06 ft^3
Containment leak rate	
0 - 24 hours	0.2 weight % per day
> 24 hours	0.1 weight % per day
Shield building volume	3.74E+05 ft^3
Shield building participation fraction	0.5
Containment leak modeling	See Table 4
Spray operation	
Time to initiate sprays	0.0 hours
Termination of sprays	0.91 hours
Recirculation spray	Not credited
Removal coefficients	
Elemental iodine	20 hr^{-1}
Particulate	4.5 hr^{-1}
Sedimentation (after spray termination)	0.1 hr^{-1}
ECCS leakage	
Containment sump volume	315,000 gal
ECCS leak rate, 0 - 30 days	12 gal/hr
Airborne percent iodine to auxiliary building	
0 - 3 hours	10%
> 3 hours	1%
ECCS leak rate to RWST	
0 - 24 hours	3 gpm
> 24 hours	1.5 gpm
Shield and auxiliary building filter efficiencies	
Elemental	95%
Organic	95%
Particulate	99%
Time until begin control room emergency HVAC	2 minutes
Normal ventilation flow rates	
Unfiltered makeup	2500 cfm (+10%)
Emergency ventilation system flow rates	
Filtered makeup	0 cfm
Filtered recirculation	2500 cfm (+10%)
Unfiltered inleakage	800 cfm

Table 7 (continued)
Assumptions Used in Radiological Consequence Analysis
Large-Break Loss-of-Coolant Accident

Control room filter efficiencies	
Elemental	90%
Organic	90%
Particulate	99%
Atmospheric dispersion factors	Table 9

Table 8
Assumptions Used in Radiological Consequence Analysis
Fuel-Handling Accident

Reactor power	1782.6 MWt
Radial peaking factor	1.7
Fission product decay period	100 hours
Number of fuel assemblies damaged	1
Fuel pool water depth	23 ft
Pool iodine effective decontamination factor	200
Fraction of fuel compliant with RG 1.183, footnote 11	0.50
Fuel gap fission product inventory (RG 1.183, footnote 11 compliant)	
I-131	8%
Kr-85	10%
Other iodines and noble gases	5%
Fraction of fuel not compliant with RG 1.183, footnote 11	0.50
Fuel gap fission product inventory (RG 1.183, footnote 11 non-compliant)	
I-131	12%
Kr-85	30%
Other iodines and noble gases	10%
Duration of release	2 hours
Time until begin control room emergency HVAC	1 minute
Normal ventilation flow rate	
Unfiltered makeup	2750 cfm
Emergency ventilation system flow rates	
Filtered makeup	0 cfm
Filtered recirculation	2500 cfm (+10%)
Unfiltered inleakage	1500 cfm
Control room filter efficiencies	
Elemental	90%
Organic	90%
Particulate	99%
Atmospheric dispersion factors	Table 9

Table 9
Atmospheric Dispersion Factors

Exclusion Area Boundary

Time (hr)	X/Q (sec/m^3)
0 - 2	2.232E-04

Low Population Zone

Time (hr)	X/Q (sec/m^3)
0 - 2	3.977E-05
2 - 24	4.100E-06
24 - 48	2.427E-06
48 - 720	4.473E-07

Control Room

Time (hr)	X/Q (sec/m^3)
0 - 8	2.93E-03
8 - 24	1.73E-03
24 - 48	6.74E-04
48 - 720	1.93E-04

ME7110 Chi-over-Q RAI status Tracking 2012-02-28.xlsx

	A	B	C	D	E	F	G	H
1	**TAC**	**Doc type**	**Source TB**	**Source TB Reviewer**	**Request by date**	**Status**	**RAI Response ML# MLnnnnnnnnn**	**Description**
2	ME7110	RAII	EICB	Alva-001	1/27/2012	firm		1, In DEK's License Amendment Request (LAR)-210, DEK proposed incorporating the control room envelope operability and surveillance requirements, R-23 operability requirements, and the control room post-accident recirculation (CRPAR) system requirements into the KPS Technical Specification (TS) ensures the systems, structures, or components (SSCs) credited for mitigating the consequences of an accident for control room occupants were included in the TS. At the same time, DEK requesting removing crediting R-23 and the control room envelope boundary from the KPS Waste Gas Decay Tank (GDT) and Volume Control (VCT) rupture accident analysis, since it determined that occupant dose consequences are achieved without crediting the control room envelope boundary or the CRPAR system. Later DEK withdrew LAR-210. However, based on the information provided in LAR-210, it is not clear why DEK in LAR-244 is requesting deleting R-23 from the TS, even though in the accident analysis performed for both LARs, DEK stated that R-23 was not credited in the proposed accident analysis. Please explain the reason to remove R-23 and replace with analysis and manual operation of the isolation dampers.
3	ME7110	RAII	EICB	Alva-002	1/27/2012	firm		2. NUREG-0737, "Clarification of TMI Action Plan Requirements," Item III.D.3.4, "Control Room Habitability Requirements," required licensees to assure that control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gas and that the plant can be safely operated or shutdown under design basis accident conditions. LAR proposed removing radiation monitor channel R-23 as a required channel for CRPAR initiation, modifying DEK previously approved by the NRC compliance with NUREG-0737. Please describe if R-23 is removed, how DEK will comply with NUREG-0737.

	A	B	C	D	E	F	G	H
1	**TAC**	**Doc type**	**Source TB**	**Source TB Reviewer**	**Request by date**	**Status**	**RAI Response ML# MLnnnnnnnnn**	**Description**
4	ME7110	RAII	EICB	Alva-003	1/27/2012	firm		3. During the NRC staff review of LAR-210, EICB issued RAI January 30, 2008 letter (ADAMS Accession No. ML080280107). DEK provided a response on its April 3, 2008 letter (ADAMS Accession No. ML080950096); the response to question 1b included a logic diagram for operation of the control room ventilation radiation monitor. To assist NRC staff review, please address the following: a. Section 3.1.1 of Attachment 1 of LAR-244 (ADAMS Accession No. ML11252A521) states that radiation monitor R-23, as a single channel, initiates both trains of the CRPAR system and each SI train initiates the associated CRPAR fan and filtration unit train. If R-23 is removed from the logic, will it be necessary that both SI trains be actuated to initiate CRPAR fans, filtration unit trains, and close dampers ACC-1A, ACC-1B, ACC-2, and ACC-5? b. This logic shows that safety injection (SI) train A closes dampers ACC-1A and ACC-1B, and SI train B closes dampers ACC-2 and ACC-5. If R-23 is removed, how will dampers ACC-2 and ACC-5 close if the SI train B actuation signal fails? c. Provide a marked logic for the control room ventilation radiation monitor assuming that R-23 is removed from the logic.
5	ME7110	RAII	EICB	Alva-004	1/27/2012	firm		4. LAR-244 is requesting removal of R-23 from the CRPAR system. Please describe how DEK would reflect removal of R-23 from the CRPAR system in an update of the FSAR for the following items: a. Figure 9.6-6, "Control Room Air Conditioning System-Flow Diagram," in the Final Safety Analysis Report (FSAR) shows R-23 location in the CRPAR system. Provide a marked diagram for an update of the FSAR after removal of R-23. b. Section 7.7.1, "Control Room," in the FSAR describes how R-23 monitors and activates the control room ventilation.
6	ME7110	RAII	EICB	Alva-005	1/27/2012	firm		5. LAR-244, Attachment 1, Section 4.2.3 and Attachment 4, Section 2.7 state that revised radiological accident analysis (RAA) credits R-23 to limit consequences of the Locked Rotor Action (LRA) and Fuel Handling Accident (FHA). However, the RAA approved in license amendment 190 (current radiological analysis of record for KPS) credited R-23 high radiation signal for mitigating the radiological consequences to control room occupants for the LRA, GDT and VCT Rupture, and FHA. Please explain why the revised RAA (submitted in LAR-244) does not state whether credit for R-23 is considered for mitigating GDT and VCT rupture.

	A	B	C	D	E	F	G	H
1	**TAC**	**Doc type**	**Source TB**	**Source TB Reviewer**	**Request by date**	**Status**	**RAI Response ML# MLnnnnnnnnn**	**Description**
7	ME7110	RAII	EICB	Alva-006	1/27/2012	firm		6. LAR-244, Section 4.2.3 describes that removal of R-23 would require manual actions to ensure post-accident control room dose is maintained within limits and are required to limit consequences of the FHA and LRA events. Note that the current accident analysis does not credit operator action to isolate the control room during for FHA. Attachment 3, Section B.3.3.7 states that manual actuation of the CRPAR System is a backup for the SI signal actuation. To assist NRC staff review, please address the following: a. Manual actuation is not part of the logic diagram for operation of the control room ventilation radiation monitor (FSAR Figure 9.6-6). Please clarify if this would be included in the logic diagram. b. SI signal is not considered for all accident events (i.e., FHA, LRA, and GDT/VCT ruptures don't consider SI). In these cases manual action would be required. Please clarify if this would be included in the logic diagram.
8	ME7110	RAII	EICB	Alva-007	1/27/2012	firm		7. LAR-244, Attachment 4, Section 2.7, 3.3.1, and 3.6.1, state that full control room isolation require action by the operator to close monitor dampers that are not included in the isolation logic (of the control room ventilation radiation monitor). This was not discussed in previous LARs or in FSARs. Please explain the following: a. Where is this information described? Provide a logic diagram and a description for operation of all dampers required for the control room ventilation radiation system. b. Are these monitor dampers closed by the SI signal? If not, what signal actuates them?
9	ME7110	RAII	AHPB	Lapin-001	2/13/2012	firm		1. Explain why it was found to be preferable to add manual actions rather than upgrade the quality classification and redundancy of Radiation Monitor R-23.

	A	B	C	D	E	F	G	H
1	**TAC**	**Doc type**	**Source TB**	**Source TB Reviewer**	**Request by date**	**Status**	**RAI Response ML# MLnnnnnnnnn**	**Description**
10	ME7110	RAII	AHPB	Lapin-002	2/13/2012	firm		2. Among the proposed changes is a change to LCO 3.9.6.a to allow the containment equipment hatch to be open during handling of recently irradiated fuel when measures are in place which ensure the capability to close equipment hatch in the event of a fuel handling accident (FHA). As described, closing the equipment hatch requires special tools and equipment, such as, a trolley, a “jactuator”, chainfalls, etc. a. How will personnel ensure that all required tools and equipment needed to close the equipment hatch are pre-staged/available and operable? b. How will personnel know whether and what kind of radiation protection equipment and clothing is needed? c. Is a written procedure required and available? d. Is training provided to all personnel who may be called upon to close the equipment hatch?
11	ME7110	RAII	AHPB	Lapin-003	2/13/2012	firm		3. A proposed new Note, applicable to LCO 3.9.6.c, would allow penetration flow paths providing direct access from the containment to outside atmosphere to be opened under administrative controls. How will each of the administrative controls be implemented: a. How will containment penetration status be communicated? (to the CR and in-plant personnel) b. How will designated personnel know their assigned penetration(s)? c. How will designated personnel be cautioned about obstructions? d. How will Operations know that designated personnel are at their posts?
12	ME7110	RAII	AHPB	Lapin-004	2/13/2012	firm		4. a. Does Radiation Monitor R-23 perform any functions other than the isolation function that is being removed? b. If yes, what functions will remain? c. If no, will all controls, displays, and logic interfaces associated with R-23 be physically removed?

	A	B	C	D	E	F	G	H
1	TAC	Doc type	Source TB	Source TB Reviewer	Request by date	Status	RAI Response ML# MLnnnnnnnnn	Description
13	ME7110	RAII	AHPB	Lapin-005	2/13/2012	firm		5. In Attachment 5 of the licensee's revised submittal, it is stated that: "Operations personnel were included in the walkdown of the control room." a. Was at least one crew included in the walkdown? b. If not, what plans are being made to validate the procedures, training, and physical interfaces with a representative sample of operators, i.e., at least one crew.
14	ME7110	RAII	AHPB	Lapin-006	2/13/2012	firm		6. The revised RAA credits manual initiation of the CRE isolation within 60 minutes of the occurrence of an LRA, and initiation of the Control Room Post Accident Recirculation (CRPAR) system within 20 minutes of occurrence of a FHA and within [60] minutes of an LRA. [In its response the licensee will clarify the differences between the current and requested licensing bases. Reviewer Lapinsky corrected the time to 60 minutes.] a. How were these completion times estimated? b. What are the actual times or the estimated required times for these actions? c. How much margin is built into the estimates of completion times?
15	ME7110	RAII	AHPB	Lapin-007	2/13/2012	firm		7.[Deleted item] For FHA and LRA, how was ALARA factored into the mitigation strategies? Will operator training address the integration of ALARA into the mitigation strategies? [This item is deleted by Reviewer Lapinski]
16	ME7110	RAII	AHPB	Lapin-008	2/13/2012	firm		8. The licensee stated in Attachment 5 of the revised submittal, that "The appropriate modifications to plant procedures will be made as part of the implementation of this amendment request. Identify all procedure changes that will be made in support of this LAR. Include the procedure numbers and titles of the affected [emergency operating] procedures. [With the additional words "emergency operating" Reviewer Lapinski clarified the scope of the requested procedures]
17	ME7110	RAII	AHPB	Lapin-009	2/13/2012	firm		9. Describe any changes to training that are necessary to support this LAR.

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1	**TAC**	**Doc type**	**Source TB**	**Source TB Reviewer**	**Request by date**	**Status**	**RAI Response ML# MLnnnnnnnnn**	**Description**
18	ME7110	RAII	SRXB	Sun-001	2/28/2012	draft		1. In an email message from Craig Sly (DEK) to Karl Feintuch (USNRC) dated September 12, 2011 12:44 PM, DEK provided some information pertaining to the percentage (%) of "Failed Fuel Following the Accident." SRXB seeks to apply information contained in the file "response 9-12-11.pdf" (one among six pdf attachments to the email of September 12, 2011 12:44, all of which are included with this message, for completeness). SRXB is providing assistance to another Technical Branch rather than using this information to prepare a safety evaluation of the requested licensing action. In your RAI Response to this item, please provide file "response 9-12-11.pdf."
19	ME7110	RAII	SRXB	Sun-002	2/28/2012	draft		2. Application Attachment 4, page 154 indicates that the actuation time of the safety injection (SI) signal (in seconds) is changed from 52.5 to 240 during rod ejection accident (REA). The reason for the change, as stated by the licensee, is that the delay of the SI signal is conservative. The Current License Basis (CLB) assumption is based on a 2-inch diameter break. The REA is specified to have a smaller 1.6 inch diameter break. The SI signal generated from a 1-inch diameter break is 240 seconds. It is not clear why a longer delay time of the actuation of the SI signal is conservative for the REA dose analysis. Please provide justification of the longer SI actuation delay time used in the REA dose analysis.
20	ME7110	RAII						
21	ME7110	RAII						

TAC	Doc type	Source Tech Branch	Source Reviewer and Ser#	Request by date	Status	ML#	Description
ME7110	RAII	EICB	Alva-001	12/29/2011	draft	MLnnnnnnnn	Assigned TAC No. This may be different than ME7110 if future sub-projects need other TAC No.
	RAII						RAII = Request for information item RAIR = Request for information response Suppl = docketed supplement
		EICB					EICB = Instrumentation and Control Branch (Alvarado) AHPB = Health Physics and Human Performance Branch (Lapinsky) AADB = Accident Dose Branch (Blumberg, Brown) ITSB = Technical Specifications Branch (no SE expected) (Hamm) SCVB = Containment and Ventilation Branch (Torres) SRXB = Reactor Systems Branch (no SE expected) (Sun, Guzzetta)
			Alva-001				Alva-nnn = Items from Reviewer Alvarado Lapin-nnn = Items from Reviewer Lapinsky Sun-nnn = Items from Reviewer Sun
				12/29/2011			Request by date (updated as mutually understood by PM, Reviewer and Licensee; maintained by PM and Licensee)
					draft		draft = as issued prior to clarification firm = as mutually understood and to be respond to by licensee resp = contains docketed response
						ML#	If blank, then = not yet docketed in ADAMS If RAIR, then = docketed ML# If RAII, then = issued RAI If Suppl, then = docketed supplement letter containing no RAIR information