April 4, 2003

Mr. Thomas Coutu
Site Vice President
Kewaunee Nuclear Power Plant
Nuclear Management Company, LLC
N490 Highway 42
Kewaunee, WI 54216-9511

SUBJECT: KEWAUNEE NUCLEAR POWER PLANT - ISSUANCE OF AMENDMENT (TAC NO. MB5718)

Dear Mr. Coutu:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 167 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant. This amendment revises the technical specifications (TSs) in response to your application dated July 26, 2002, as supplemented February 27, March 14, March 19, March 21 (2 letters), and April 3, 2003.

The amendment revises TSs for use of Westinghouse 422 VANTAGE + nuclear fuel with PERFORMANCE + features.

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. 167 to
License No. DPR-43
2. Safety Evaluation

cc w/encls: See next page

April 4, 2003

Distribution w/encls:

GHill (2)	JLamb
PUBLIC	WBeckner, TSB
PD 3-1 r/f	ACRS
OGC	GGrant, RIII
THarris	

Mr. Thomas Coutu
Site Vice President
Kewaunee Nuclear Power Plant
Nuclear Management Company, LLC
N490 Highway 42
Kewaunee, WI 54216-9511

SUBJECT: KEWAUNEE NUCLEAR POWER PLANT - ISSUANCE OF AMENDMENT (TAC NO. MB5718)

Dear Mr. Coutu:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 167 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant. This amendment revises the technical specifications (TSs) in response to your application dated July 26, 2002, as supplemented February 27, March 14, March 19, March 21 (2 letters), and April 3, 2003.

The amendment revises TSs for use of Westinghouse 422 VANTAGE + nuclear fuel with PERFORMANCE + features.

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. 167 to
License No. DPR-43
2. Safety Evaluation
cc w/encls: See next page *See memo **See previous concurrence

ADAMS ACCESSION NUMBER: ML030940276

OFFICE	**PM:PD3-1**	**LA:PD3-1**	**SC:SRXB**	**SC:EEIB**	**SC:SPSB**	**OGC**	**SC:PD3-1**
NAME	**JLamb**	**THarris****	**FAkstulewicz***	**EMarinos***	**FReinhart***	**RWeisman****	**LRaghavan**
DATE	**04/04/03**	**3/31/03**	**4/01/03**	**2/24/03**	**3/21/03**	**4/03/03**	**04/04/03**

OFFICIAL RECORD COPY

Kewaunee Nuclear Power Plant

cc:

John Paul Cowan
Chief Nuclear Officer
Nuclear Management Company, LLC
27780 Blue Star Memorial Highway
Cover, MI 49043

Kyle Hoops
Plant Manager
Kewaunee Nuclear Power Plant
N490 Highway 42
Kewaunee, WI 54216-9511

Gordon P. Arent
Manager, Regulatory Affairs
Kewaunee Nuclear Power Plant
N490 Highway 42
Kewaunee, WI 54216-9511

David Molzahn
Nuclear Asset Manager
Wisconsin Public Service Corporation
600 N. Adams Street
Green Bay, WI 54307-9002

Thomas Webb
Nuclear Asset Manager
Wisconsin Public Service Corporation
600 N. Adams Street
Green Bay, WI 54307-9002

Resident Inspectors Office
U. S. Nuclear Regulatory Commission
N490 Hwy 42
Kewaunee, WI 54216-9510

Regional Administrator
Region III
U. S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60532-4351

Jonathan Rogoff, Esquire
General Counsel,
Nuclear Management Company, LLC
700 First Street
Hudson, WI 54016

Larry L. Weyers
Chairman, President and CEO
Wisconsin Public Service Corporation
600 North Adams Street
Greey Bay, WI 54307-9002

David Zellner
Chairman - Town of Carlton
N2164 County B
Kewaunee, WI 54216

Sarah Jenkins
Electric Division
Public Service Commission of Wisconsin
PO Box 7854
Madison, WI 53707-7854

NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-305

KEWAUNEE NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 167
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 July 26, 2002, as supplemented February 27, March 14, March 19, March 21 (2 letters), and April 3, 2003, 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. 167, are hereby incorporated in the license. The licensee's shall operate the facility in accordance with the Technical Specifications.

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

Attachment: Changes to the Technical Specifications

Date of Issuance: April 4, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 167

FACILITY OPERATING LICENSE NO. DPR-43

DOCKET NO. 50-305

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 ii	TS ii
TS iv	TS iv
TS vi	TS vi
TS 1.0-6	TS 1.0-6
TS 2.1-1	TS 2.1-1
TS 2.3-2 through TS 2.3-3	TS 2.3-2 through TS 2.3-3
TS 3.1-7	TS 3.1-7
TS 3.4-1 through 3.4-3	TS 3.4-1 through 3.4-3
TS 3.10-1 through TS 3.10-3	TS 3.10-1 through TS 3.10-3
TS Figure 3.1-3	TS Figure 3.1-3
TS 3.10-7	TS 3.10-7
Table TS 4.1-1	Table TS 4.1-1
TS 5.2-1	TS 5.2-1
TS 5.3-1	TS 5.3-1
TS 6.9-4 through 6.9-6	TS 6.9-4 through 6.9-6

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 167 TO FACILITY OPERATING LICENSE NO. DPR-43

NUCLEAR MANAGEMENT COMPANY, LLC

KEWAUNEE NUCLEAR POWER PLANT

DOCKET NO. 50-305

TABLE OF CONTENTS

TABLE OF CONTENTS

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 167 TO FACILITY OPERATING LICENSE NO. DPR-43

NUCLEAR MANAGEMENT COMPANY, LLC

KEWAUNEE NUCLEAR POWER PLANT

DOCKET NO. 50-305

1.0 INTRODUCTION

By a letter dated July 26, 2002, as supplemented February 27, March 14, March 19, March 21, and April 3, 2003, Nuclear Management Company, LLC (NMC or the licensee) requested a license amendment to revise the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TSs). The proposed changes would revise the TSs to accommodate the use of Westinghouse 422 VANTAGE + nuclear fuel with PERFORMANCE + features (422V+ fuel). The 422V+ fuel is currently loaded in the KNPP reactor core as part of the lead test assembly program for Cycle 25. During this and succeeding refueling outages, all discharged fuel assemblies will be replaced with fuel assemblies containing 422V+ fuel.

The licensee has analyzed or evaluated the nuclear and fuel design, thermal-hydraulic design, and loss-of-coolant accident (LOCA) and non-LOCA transient and accident analyses in support of the proposed TSs changes. The licensee has assumed a rated core power of 1772 megawatts thermal (MWt), which is 107.4 percent of the current rated core power of 1650 MWt, in many of the supporting safety analyses. The licensee intends to utilize these safety analyses to support a pending application for a 1.4 percent measurement uncertainty recapture (MUR) power uprate and a planned six percent uprate license amendment request (LAR). Although the licensee is not requesting a power uprate with this proposed amendment, the U.S. Nuclear Regulatory Commission (NRC) staff's safety evaluation report (SER) discusses the adequacy of most transient and accident analyses at a rated thermal power of 1772 MWt. However, this safety evaluation does not authorize operation above the current licensed power level of 1650 MWt.

Additionally, the licensee proposed to relocate from the TS to the Core Operating Limits Report (COLR) the overtemperature delta-temperature (OTΔT) and overpower delta-temperature (OPΔT) parameter constants (τ_1, τ_2, τ_3, K_1,K_2,K_3,K_4,K_5,K_6, T', and P') and the slope and breakpoint values for the delta-flux function, f(ΔI). The relocation of these values from the TS to the COLR is supported by NRC guidance provided in Generic Letter 88-16 (GL 88-16), "Removal of Cycle-Specific Parameter Limits from Technical Specifications," and WCAP-14483-A, "Generic Methodology For Expanded Core Operation Limits Report."

ENCLOSURE

The supplemental information dated February 27, March 14, March 19, March 21, and April 3, 2003, contained clarifying information, did not change the scope of the July 26, 2002, application or the initial no significant hazards consideration determination, and did not expand the scope of the original *Federal Register* notice.

2.0 EVALUATION

The NRC staff has reviewed the licensee's submittals and analyses related to the implementation of the 422V+ fuel. The NRC staff's review focused on the nuclear and fuel design, thermal-hydraulic design, LOCA and non-LOCA transient and accident analyses, and instrument uncertainties. Each of these subjects is evaluated separately in the respective sections which follow.

The U.S. Atomic Energy Commission (AEC) issued a "Safety Evaluation of the Kewaunee Nuclear Power Plant" on July 24, 1972, supplemented December 18, 1972, and May 10, 1973. KNPP is not a General Design Criteria (GDC) plant. However, the AEC performed a technical review of the KNPP against the GDC in effect at the time and concluded that the KNPP design generally conforms to the GDC; therefore, the NRC reviewed proposed changes using the GDCs as regulatory acceptance criteria where applicable.

2.1 Fuel System Design Evaluation

2.1.A Fuel System Design Regulatory Evaluation

NUREG-0800, "Standard Review Plan," (SRP) Section 4.2 provides guidance for the NRC staff's review regarding fuel system design. The objectives of the fuel system review are to provide assurance that:

(a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs),

(b) fuel system damage is never so severe as to prevent control rod insertion when it is required,

(c) the number of fuel rod failures is not underestimated for postulated accidents, and

(d) 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 designing the reactivity control system to control reactivity changes reliably, in conjunction with the ECCS, with appropriate margin for stuck rods, and assure that the capability to cool the core is maintained under postulated 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.

2.1.B Fuel System Design Technical Evaluation

The licensee's evaluation of the fuel system design is documented in Attachment 4, Section 2 of the licensee's submittal (Reference 1). The licensee evaluated the proposed Westinghouse 422V+ fuel 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 NRC staff approved the use of Westinghouse 422V+ lead test assemblies for the current KNPP operating Cycle 25, and similar fuel assemblies are in their second cycle of operation in Point Beach Units 1 and 2.

The proposed 422V+ fuel design incorporates Westinghouse VANTAGE+ and PERFORMANCE+ fuel features. VANTAGE+ features include ZIRLO cladding, ZIRLO guide thimbles and instrument tubes, annular or solid pellets in the top and bottom axial blankets and modifications for high burnup design. The NRC staff previously approved these VANTAGE+ design features as documented in WCAP-12610-P-A (Reference 4). The NRC staff SER for WCAP-12610-P-A included a condition for plant specific applicability that, "[f]uture plant applications of VANTAGE+ design must also demonstrate that the ZIRLO oxidation data is applicable for these applications." The licensee satisfied this condition by stating that KNPP's application is consistent with other plants in the Westinghouse corrosion database, and that the proposed KNPP 422V+ fuel meets the Westinghouse internal oxidation limit of < 100 microns (Reference 2, Attachment 3 Request for Additional Information (RAI), Question 1). The NRC staff reviewed the licensee's application regarding the plant-specific conditions of WCAP-12610-P-A, and the NRC staff finds it acceptable because the licensee meets the conditions of the NRC-approved WCAP-12610-P-A and the oxidation acceptance criterion of less than 100 microns.

PERFORMANCE+ features include low cobalt top and bottom nozzles, Zirconium oxide coated lower fuel rod, ZIRLO mid-grids, and mid-enrichment of the annular or solid pellets in axial blankets. The PERFORMANCE+ features were developed and reviewed by Westinghouse in SECL-92-305 (Reference 5) under 10 CFR 50.59 guidelines (Reference 2, Attachment 3 RAI, Question 6). Other significant mechanical features of the 422V+ fuel design include a 0.422 inch outer diameter (OD) fuel rod and instrumentation tube, and a new optimized fuel assembly style mid-grid. Westinghouse developed and licensed the new mid-grid design under the guidelines of the FCEP process (Reference 6).

The licensee evaluated the compatibility of the Westinghouse 422V+ and the current Framatome/ANP fuels and concluded that the two fuel types are hydraulically and mechanically compatible. The licensee demonstrated compatibility of the fuel assemblies, including fuel rods, grid spacers, guide thimble and instrumentation tubes, reconstitutable top nozzle and debris filter bottom nozzle. The hydraulic compatibility determination is based on experience-based parameter comparisons of component loss coefficients, rod bundle geometry and grid axial elevations. The hydraulic compatibility conclusion was supported by the fuel assembly crossflow results (Reference 2, Attachment 3 RAI, Questions 7 and 23.a) and fuel assembly lift forces (Reference 2, Attachment 3 RAI, Questions 7 and 23.b). The licensee's analyses demonstrated that no significant crossflow induced vibration will occur in the transition core and that fuel assembly lift forces are acceptable. Additionally, the Westinghouse design criterion requires that there should be outer grid strap overlap between any two fuel assemblies in the core throughout their life in the core; Westinghouse design criterion has been satisfied (Reference 2, Attachment 3 RAI, Question 8). This criterion is used to ensure crossflow and rod vibration are minimized. Based on the licensee's submittals, the NRC staff finds that all necessary criteria are satisfied such that the two fuel types are shown to be hydraulically compatible.

The mechanical compatibility evaluation consisted of a geometric evaluation of grid overlap and anti-snag features. Acceptability of the Westinghouse fuel and loads imparted on co-resident fuel during seismic LOCA events was also evaluated (Reference 2, Attachment 3 RAI, Questions 13 and 14). The licensee applied a Leak Before Break methodology to determine which LOCA events to consider, and applied the square-root-of-sum-of-squares (SRSS) method identified in Appendix A to SRP 4.2 to demonstrate that the combined loadings are acceptable. The NRC staff previously approved this application of LBB methodology for KNPP (Reference 2, Attachment 3 RAI, questions 13). For the homogeneous 422V+ core, the licensee determined that all allowable grid load limits are satisfied. The licensee evaluated two limiting mixed core configurations and demonstrated adequate grid load margin for all fuel assemblies except for up to three grid spacers on the periphery of the core in the limiting mixed core condition. The licensee determined that the damaged grids would not result in fuel rod mechanical fracture. Thus, no fuel failure is predicted in these fuel assemblies. The licensee's analyses demonstrated that a coolable core geometry is maintained. The licensee utilized the WEGAP, WECAN and NKMODE computer codes to perform these analyses (Reference 2, Attachment 3 RAI, question 14). WEGAP has been previously approved by the NRC as documented in the SER for WCAP-9401-P-A (Reference 12). Westinghouse has conducted benchmark calculations for all three codes, and all are governed by Westinghouse software control, which requires validation and verification, and Westinghouse Quality Management System (QMS) which is 10 CFR Part 50, Appendix B certified and has been audited and approved by the NRC.

The licensee evaluated 422V+ fuel rod performance under operating conditions consistent with a core power of 1772 MWt and a peaking factor $F^N_{\Delta H}$ limit of 1.70. The Westinghouse fuel rod design criteria were utilized to evaluate fuel rod performance. The Westinghouse criteria capture all criteria of SRP 4.2 and also include additional criteria (see Reference 2, Attachment 3 RAI, Question 10). The licensee used NRC-approved models (References 4, 7 and 8) and NRC-approved methods (References 9 and 10) and has demonstrated that all fuel rod design criteria are satisfied. The licensee used an NRC approved methodology (Addendum 2 to WCAP-12488) to evaluate the clad stress criteria and has demonstrated that the acceptance criteria are satisfied (Reference 34). The licensee utilized the NRC-approved Performance, Analysis and Design (PAD) 3.4 and PAD 4.0 codes with NRC-approved models (References 7 and 8) for in-reactor behavior to calculate the fuel rod performance for the fuel rod design criteria over its entire irradiation history.

The licensee evaluated the Westinghouse 422V+ fuel design for impacts on rod cluster control assembly (RCCA) insertion. 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 thermal power of 1772 MWt. 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 (Reference 2, Attachment 3 RAI, Question 9). Therefore, RCCA drop times are acceptable.

2.1.C Fuel System Design Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed fuel transition on the fuel system design. For the reasons set forth above, the NRC staff concludes that the licensee has adequately accounted for the effects of the proposed fuel transition on the fuel system and demonstrated that (1) the fuel system will not be damaged as a result of normal operation and AOOs, (2) 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 proposed fuel transition. Therefore, the NRC staff finds the proposed fuel transition acceptable with respect to the fuel system design.

2.2 Nuclear Design Evaluation

2.2.A Nuclear Design Regulatory 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 reactor coolant pressure boundary (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 worths, 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 designing the core so that the prompt inherent nuclear feedback characteristics tend to 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 providing instrumentation and controls to monitor variables and systems affecting the fission process over anticipated ranges for normal operation, AOOs; and accident conditions, and for maintaining the variables and systems within prescribed operating ranges; (5) GDC-20 for designing the reactivity control systems for automatic initiation to assure that acceptable fuel design limits 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 assuring that a single malfunction of the reactivity control system does not result in exceeding the SAFDLs; (7) GDC-26 for providing two independent reactivity control systems of different design, with each system having the capability to control the rate of reactivity changes resulting from planned, normal power changes; (8) GDC-27 for designing the reactivity control system to control reactivity changes reliably, in conjunction with the ECCS, with appropriate margin for stuck rods, and assure that the capability to cool the core is maintained under postulated accident conditions; and (9) GDC-28 for assuring that the effects of postulated reactivity accidents neither result in damage to the RCPB greater than limited local yielding, nor cause sufficient damage to impair significantly the capability to cool the core. Specific review criteria are contained in SRP Section 4.3

2.2.B Nuclear Design Technical Evaluation

The licensee evaluated the KNPP nuclear design bases and methodologies for the use of Westinghouse 422V+ fuel with PERFORMANCE+ features considering a core power level of 1772 MWt. The Westinghouse 422V+ fuel design was evaluated based on a lead rod burnup of up to 75 gigawatt days per metric ton uranium (GWD/MTU). Generating nuclear parameters and designing the fuel assemblies to 75 GWD/MTU is accomplished by considering mechanical aspects of the fuel such as irradiation growth and sizing of the fuel assembly/fuel rod (Reference 2, Attachment 3 RAI, Question 17). Currently, the licensee's Facility Operating License DPR-43, paragraph 2.C.(5) limits the maximum rod average burnup for any rod to 60 GWD/MTU. Accordingly, operation with the 422V+ fuel will be limited to the current KNPP TS licensed fuel burnup limit of 60 GWD/MTU; nonetheless, extension to 62 GWD/MTU may be possible based on Appendix R of WCAP-12488-A (Reference 6). Two design features of the 422V+ fuel which are not present in the current KNPP Framatome/ANP fuel include: (1) the use of ZIRLO material for fuel cladding, guide thimble tubes, instrumentation tubes, and LPD mid-grids, and (2) a reduced fuel stack height of 0.75 inches within the assembly. Use of the ZIRLO alloy allows additional flexibility in fuel management provided by increased lead rod burnups. The 422V+ fuel stack height reduction accommodates fission gas release for the extended burnup design.

The licensee performed reload transition core analyses to demonstrate that the nuclear design is acceptable and satisfies the acceptance criteria of SRP Section 4.3. The licensee performed all applicable analyses using NRC-approved methods and computer codes, and demonstrated that all applicable Westinghouse design limits and acceptance criteria are satisfied and all applicable TS limits will be satisfied for the proposed 422V+ fuel design (Reference 2, Attachment 3 RAI, Question 20). Standard NRC-approved Westinghouse reload design philosophy which evaluates the reload core key safety parameters such as power distributions, peaking factors, rod worths and reactivity parameters was applied (Reference 13). Standard Westinghouse nuclear design analytical methods and models (References 13 -15) were applied to evaluate the neutronic behavior of both the Westinghouse 422V+ fuel and the current Framatome/ANP fuel through the transition cores. The licensee provided justification for the application of this Westinghouse neutronics methodology to the current Framatome/ANP fuel by demonstrating the nature and magnitude of the differences between the Framatome/ANP and 422V+ fuel designs is significantly less than the differences between the various Westinghouse fuel designs (Reference 2, Attachment 3 RAI, Question 15). Additionally, the licensee verified that all applicable code and model limitations and restrictions were satisfied in performing these analyses (Reference 2, Attachment 3 RAI, Question 35).

The licensee developed three representative core models for performing the transition core analyses (Reference 2, Attachment 3 RAI, Question 18). The first transition cycle model is used to capture the initial and predominant transition core effects. A second transition cycle model and a third, all Westinghouse 422V+, core model were developed to capture the core characteristics through the all 422V+ loaded core. The loading patterns were developed based on projected energy requirements of approximately 500 effective full-power days for KNPP. The licensee's safety analyses support a maximum nuclear enthalpy rise hot channel factor ($F^N_{\Delta H}$) limit of 1.70, and a total peaking factor (F_Q) limit of 2.50. These models were developed to demonstrate that adequate margin exists between typical safety parameter values and the corresponding limits to allow flexibility in designing actual reload cores. Cycle specific core reload design analyses will continue to verify the acceptability of future designed 422V+ core designs for KNPP.

2.2.C Nuclear Design Conclusion

The NRC staff has reviewed the licensee's analyses related to the effect of the proposed fuel transition on the nuclear design. For the reasons set forth above, the NRC staff concludes that the licensee has adequately accounted for the effects of the proposed fuel transition 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. Therefore, the NRC staff finds the proposed fuel transition acceptable with respect to the nuclear design.

2.3 Thermal and Hydraulic Design Evaluation

2.3.A Thermal and Hydraulic Design Regulatory Evaluation

The NRC staff reviews the thermal and hydraulic design of the core and the reactor coolant system (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.

2.3.B Thermal and Hydraulic Design Technical Evaluation

To support the fuel transition, the licensee performed extensive core thermal-hydraulic (T-H) analyses. The analyses evaluated the proposed Westinghouse 422V+ fuel with respect to departure from nucleate boiling (DNB) performance, hydraulic compatibility of the current Framatome and proposed Westinghouse fuel types and T-H related transition core effects. Specific fuel related design differences evaluated include:

- Decrease from a 0.424 inch OD fuel rod to a 0.422 inch OD fuel rod
- Change from 14x14 Zircaloy-4 mid-grid to 14x14 ZIRLO mid-grid
- Decrease from 0.424 inch OD instrumentation tubes to 0.422 inch OD instrumentation tubes
- Decrease from 0.541 inch guide tubes to 0.526 inch guide tubes

The licensee's DNB analyses for the Westinghouse 14x14 422V+ fuel incorporate the Westinghouse Revised Thermal Design Procedure (RTDP) (Reference 16), the WRB-1 DNB correlation (Reference 17) and the Westinghouse VIPRE code (Reference 18). The licensee's current T-H analytical method, consisting of the Standard Thermal Design Procedure (STDP) and the W-3 DNB correlation, will continue to be used when parameters fall outside the range bounded by the WRB-1 correlation. The NRC staff previously approved the application of RTDP methodology and the WRB-1 DNB correlation for Westinghouse 422V+ fuel design for a similar LAR for Point Beach (Reference 19).

The WRB-1 DNB correlation is based entirely on rod bundle experimental data. The NRC staff has previously approved a 95/95 correlation limit (the probability that DNB will not occur on the most limiting fuel rod is at least 95 percent, at a 95 percent confidence level) of 1.17 for both Westinghouse 14x14 Optimized Fuel Assembly (OFA) and 15x15 OFA fuel designs (Reference 2, Attachment 3 RAI, Question 21). The WRB-1 DNB correlation and an associated correlation limit of 1.17 has also been shown to be acceptable for use with the Westinghouse VIPRE code (Reference 18), and has previously been approved for the Westinghouse 14x14 422V+ fuel design at Point Beach (Reference 19). The licensee demonstrated that the new mid-grid meets all design criteria of existing tested mid-grids that form the basis of the WRB-1 correlation database and that the WRB-1 correlation with a 95/95 correlation limit of 1.17 applies to the new mid-grid (Reference 21). Accordingly, the NRC staff finds that the WRB-1 correlation and a correlation limit of 1.17 is applicable for the proposed Westinghouse 14x14 422V+ fuel design to be used at KNPP.

The RTDP methodology statistically combines uncertainties associated with plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes and DNB correlation predictions to obtain an overall DNB uncertainty factor. This factor is used to define the design limit departure from nucleate boiling ratio (DNBR) value which ensures that the correlation limit is satisfied. In this methodology, the plant safety analyses are performed using input parameters at their nominal values, because the parameter uncertainties are included in the design limit DNB value. KNPP specific instrumentation uncertainties are documented in the KNPP RTDP Instrument Uncertainty Methodology Report (Reference 20) and are calculated at a power level of 1780 MWt (NSSS power)(Reference 2, Attachment 3 RAI, Question 3). The licensee provided information which addressed how each of the seven conditions specified in the NRC staff's SER for the RTDP methodology are satisfied (Reference 2, Attachment 3 RAI, Question 5) for its application at KNPP. The NRC staff reviewed the licensee's application and how the licensee addressed each of the seven conditions specified in the NRC staff's SER for the RTDP methodology; the NRC staff finds that the licensee has satisfied each of the conditions and established the applicability of the methodology to KNPP.

The licensee used the NRC-approved RTDP methodology (Reference 16) and calculated design limit DNBR values for the Westinghouse 422V+ fuel to be 1.24/1.24 for typical/thimble cells. For performing safety analyses, the design limit DNBR values are conservatively increased to provide DNBR margin to offset the effects of rod bow, transition core penalties and any other DNBR penalties that occur, and to provide flexibility in the design and operation of the plant. This increased DNBR limit used in evaluating the updated safety analysis report (USAR) Chapter 14 transients is termed the safety analysis limit. The licensee calculated the KNPP 422V+ fuel safety analysis DNBR limit to be 1.34/1.34 (typical/thimble). The licensee has retained 7.46 percent DNBR margin between the design limit and the safety analysis limit DNBR values and determined that this retained DNBR margin is sufficient for operation at a thermal power level of 1673 MWt. The NRC staff reviewed the licensee's use of the RTDP methodology and the calculated DNBR values and finds that the licensee applied the RTDP methodology at KNPP using appropriate plant-specific inputs. Therefore, the NRC staff concludes that the calculated DNBR values acceptable.

The licensee evaluated the hydraulic compatibility of the Westinghouse 14x14 422V+ fuel and the current 14x14 Framatome/ANP fuel designs. The hydraulic compatibility determination is based on experience based parameter comparisons of component loss coefficients, rod bundle geometry and grid axial elevations. The hydraulic compatibility conclusion was supported by the fuel assembly crossflow results. The licensee's analyses demonstrated that the fuel assembly crossflow that exists for the transition core is well within the bounding Westinghouse experience basis of transition core analyses (Reference 2, Attachment 3 RAI, Questions 7 and 23.a).

Additionally, the licensee performed full-scale hydraulic tests on the 14x14 Westinghouse 422V+ fuel assembly design to confirm pressure drop compatibility with the current Framatome/ANP 14x14 fuel design. The 14x14 Westinghouse 422V+ fuel assembly design overall pressure loss is approximately 10 percent larger than that for the current Framatome/ANP 14x14 fuel. The licensee determined that for the transition cores, this difference in pressure drop equates to a maximum co-resident flow increase of 4.88 percent in the Framatome/ANP assemblies and a 10 percent increase in liftoff forces. This increase in flow for the Framatome/ANP fuel assemblies reduces the margin to the assembly liftoff force limit (and increases the margin in the Westinghouse assemblies) and increases the margin to the DNBR (and decreases the margin in the Westinghouse assemblies).

The licensee performed detailed liftoff force calculations and determined that the liftoff force margin under normal and transient operating conditions for the Framatome/ANP fuel is acceptable for the first transition core operating cycle. The licensee evaluated the decrease in margin to DNBR for the Westinghouse fuel assemblies and found this to be acceptable based on the detailed DNB analyses performed as part of the safety analyses and verified in the reload safety evaluation. The decrease in margin to DNBR for the proposed Westinghouse fuel will be regained when the transition to all Westinghouse fuel is complete. The NRC staff reviewed the hydraulic compatibility of the 422V+ fuel and agrees with the licensee's assessment.

2.3.B Thermal and Hydraulic Design Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed fuel transition on the thermal and hydraulic design of the core and the RCS. For the reasons set forth above, the NRC staff concludes that the licensee has adequately accounted for the effects of the proposed fuel transition at a power level of 1673 MWt on the thermal and hydraulic design, and demonstrated that the design has been accomplished using acceptable analytical methods, is equivalent to proven designs, provides acceptable margins of safety from conditions which would lead to fuel damage during normal reactor operation and AOOs, and 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 fuel transition. Therefore, the NRC staff finds the proposed fuel transition acceptable with respect to thermal and hydraulic design.

2.4 Transient and Accident Analyses

The licensee reanalyzed the LOCA and non-LOCA transients and accidents in support of the transition to 422V+ fuel. These analyses were performed at a rated core power of 1772 MWt (which is 107.4 percent of the current rated power of 1650 MWt) and incorporated plant parameter values for those operating conditions. The NRC staff's review of the LOCA and non-LOCA transients and accidents is discussed in the following sections.

2.4.1 LOCA Analyses

2.4.1.1 Large-Break LOCA and Small-Break LOCA Analyses

2.4.1.1.A Large-Break LOCA and Small-Break LOCA Regulatory Evaluation

The NRC's acceptance criteria are based on (1) 10 CFR 50.46 and Appendix K to 10 CFR Part 50 for using an acceptable evaluation model for LOCA analyses and for providing ECCS equipment that refills the vessel in a timely manner for a LOCA; (2) GDC-4 for protecting structures, systems, components important to safety against the dynamic effects associated with flow instabilites and loads; (3) GDC-27 for designing the reactivity control system to control reactivity changes reliably, in conjunction with the ECCS, with appropriate margin for stuck rods, and assure that the capability to cool the core is maintained under postulated accident conditions; and (4) GDC-35 for providing abundant emergency core cooling to transfer heat from the core following any loss of reactor coolant at a rate so that fuel and clad damage will not interfere with continued effective core cooling. Specific review criteria are contained in SRP Sections 6.3 and 15.6.5.

2.4.1.1.B Large-Break LOCA and Small-Break LOCA Technical Evaluation

The licensee performed KNPP large-break loss-of-coolant accident (LBLOCA) and small-break loss-of-coolant accident (SBLOCA) reanalyses for cores containing Westinghouse 422V + (ZIRLO clad) fuel and Framatome 14x14 (Zircaloy clad) fuel assuming 102 percent (1807 MWt) of a core-rated power of 1772 Mwt. This power (1772 Mwt) represents a 7.4 percent increase over the present KNPP rated core power of 1650 Mwt. The licensee has also indicated that it will incorporate Westinghouse 422V+ (ZIRLO-clad) fuel into upcoming KNPP cores. For at least the first operating cycle with the 422V+ fuel, Kewaunee will be operated with a mixed core configuration with Framatome fuel .

WCAP-12610-P-A, April 1995 (Ref. 4), describes 422V+ fuel, and Appendices F and G to that document describe performing LOCA analyses for cores containing 422V+ fuel using Westinghouse SBLOCA and LBLOCA methodologies. The licensee used the Westinghouse best estimate LBLOCA analysis methodology described in NRC-approved WCAP-14449-P-A, Revision 1, October 1999 (Ref. 36), to perform KNPP LBLOCA analyses. This methodology is specifically applicable to Westinghouse pressurized-water reactors (PWRs) designed with upper plenum injection (UPI), including KNPP. The methodology also applies to this class of plants for LBLOCA analysis of 422V+ fuel (Ref. 37). The licensee used the Westinghouse "COSI" SBLOCA evaluation (Appendix K) methodology described in NRC-approved WCAP-10054-P-A, Addendum 2, Rev. 1, July 1997 (Ref. 38). 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 are as required by 10 CFR 50.46(a)(3).

The NRC staff reviewed the licensee's analytical LOCA methodologies and licensing analysis results for conformance with applicable requirements of 10 CFR 50.46 (Ref. 39). The NRC staff also used GDC-35 (Ref 40.) and 10 CFR Part 50, Appendix K (Ref. 41) in this review.

In response to an NRC staff question, in a letter dated February 27, 2003, (Ref. 2) the licensee confirmed that both the Westinghouse LOCA methodologies described above apply specifically to KNPP by providing a statement that NMC and Westinghouse have ongoing processes which assure that the values and ranges of the LOCA analysis inputs for peak cladding temperature-sensitive parameters bound the values and ranges of the as-operated plant for those parameters.

The NRC evaluation of the previously-approved WCAP-12610-P-A and its Appendices F and G, concluded that the Westinghouse SBLOCA and LBLOCA methodologies that the NRC staff has approved are suitable for analyses of 422V+ fuel. The NRC staff also concluded that "because of the close similarity between Vantage+ and Vantage-5 (Zircaloy-clad) fuel assemblies a mixed core penalty need not be applied to any mixed core combination of Vantage-5 and Vantage+ fuel assemblies, if both have the same design features." The NRC has understood that the most significant of these "design features" include geometry (e.g. differences in spacer or mixing grids) and surface roughness. In a letter dated February 27, 2003, the licensee supplemented its proposal to include additional information regarding the treatment of mixed cores in the LOCA analyses. In its LBLOCA analyses, the licensee explicitly accounted for the mixed core by established techniques and calculated the performance for both fuels.

For the LOCA analyses, the licensee showed that the 422V+ fuel was bounding for peak clad temperature (PCT) over the Framatome fuel, because (1) geometry differs between the two, (2) the 422V+ fuel is fresh fuel versus at least one cycle burn-up of the resident Framatome fuel, (3) the 422V+ fuel will be located in the highest powered regions, and (4) the 422 V+ fuel has higher peaking.

In its February 27, 2003, letter, the licensee stated that the calculated post-LOCA oxidation for the 422V+ fuel bounds that calculated for the Framatome fuel, because the calculated PCTs for the 422V+ fuel are higher than for the Framatome fuel for like LOCA events, and cladding temperature strongly affects the oxidation rate. However, at the NRC staff's request, the licensee also addressed the concern that the resident Framatome fuel may have pre-existing oxidation that must be considered. For SBLOCA scenarios, 422V+ fuel post-LOCA oxidation was very low, less than 0.1 percent, because of the very low calculated PCTs for SBLOCAs of 1030 ^{o}F. Since the post-SBLOCA oxidation was so low, the LBLOCA oxidation bounded the SBLOCA oxidation for both fuels.

The LBLOCA calculations produced the following results:

- For the 422V+ fuel, the calculated 95th percentile PCT is less than (<)2084 °F, the calculated maximum local cladding oxidation is <8.44 percent, and the maximum core-wide hydrogen generation is less than 0.74 percent for all fuel.

- For the resident Framatome fuel, the calculated 95th percentile PCT is <1829 °F, the calculated maximum local cladding oxidation is <4.0 percent, and the maximum core-wide hydrogen generation is less than 0.74 percent for all fuel.

The NRC staff notes that the pre-existing oxidation for the Westinghouse fuel is negligibly small because it is fresh fuel. The licensee stated in its March 19, 2003, letter, that the calculated pre-transient oxidation for the Framatome fuel through the final cycle is less than 10.6 percent equivalent cladding reacted. Therefore, the sum of the pre-transient and post-LOCA oxidation for the Framatome fuel is less than 14.6 percent. The licensee also pointed out that the fuel with the highest LOCA oxidation is not the same fuel as that having the highest pre-accident oxidation. Therefore, the licensee concluded that the maximum total oxidation is less than the sum of the pre-accident oxidation and the LOCA oxidation.

The NRC staff also notes that the pre-existing oxidation of the Framatome fuel is not expected to contribute to the LOCA maximum core-wide hydrogen generation.

The SBLOCA calculations produced the following results:

- For the 422V+ fuel, the calculated PCT is 1030 °F for a 3-inch break, the calculated maximum local cladding oxidation is <0.1 percent, and the maximum core-wide hydrogen generation is less than 1.0 percent for all fuel. As stated above, the pre-existing oxidation of the Framatome fuel is not expected to contribute to the LOCA maximum core-wide hydrogen generation.

These low values can be attributed to the high high-pressure injection to reactor power ratio for the KNPP.

The 422V+ fuel calculated SBLOCA PCT of 1030 °F and oxidation values less than 0.1 percent bound the values of the resident Framatome fuel. Since the SBLOCA PCT and oxidation values are so low for the bounding 422V+ fuel, the NRC staff did not require the licensee to perform SBLOCA analyses for the Framatome fuel.

While the pre-existing oxidation of the resident Framatome fuel bounds the SBLOCA oxidation, the total LBLOCA oxidation bounds the SBLOCA oxidation. The maximum core-wide hydrogen generation for SBLOCA is less than 1.0 percent for all fuel.

The licensee calculated LBLOCA and SBLOCA bounding values for PCT (2084 °F, and 1030 °F, respectively), oxidation (less than 14.6 percent and 0.1 percent, respectively), and core wide hydrogen generation less than 1.0 percent. The licensee concludes that these calculated results for PCT, oxidation, and core-wide hydrogen generation are less than the limits of 2200 °F, 17 percent, and 1.0 percent specified in 10 CFR 50.46(b)(1)-(3) respectively.

As discussed above, the licensee has performed LBLOCA and SBLOCA analyses for KNPP at a power of 1772 MWt using approved Westinghouse methodologies. The licensee has shown that these methodologies apply specifically to the KNPP.

As discussed above, the licensee's LBLOCA and SBLOCA calculations demonstrated the following:

The licensee's calculated LBLOCA and SBLOCA values for PCT, oxidation, and core-wide hydrogen generation are less than the limits of 2200 ^{o}F, 17 percent, and 1.0 percent specified in 10 CFR 50.46(b)(1)-(3) respectively.

2.4.1.1.C Large-Break LOCA and Small-Break LOCA Conclusion

As stated above in Section 2.1.B, the licensee's analyses demonstrated that a coolable core geometry is maintained. The licensee utilized the WEGAP, WECAN and NKMODE computer codes to perform these analyses (Reference 2, Attachment 3 RAI, question 14). WEGAP has been previously approved by the NRC as documented in the SER for WCAP-9401-P-A (Reference 12). Westinghouse has conducted benchmark calculations for all three codes, and all are governed by Westinghouse software control, which requires validation and verification, and Westinghouse Quality Management System (QMS) which is 10 CFR Part 50, Appendix B certified and has been audited and approved by the NRC. Therefore, the NRC staff concludes that the licensee's LOCA analyses are acceptable and demonstrate that the KNPP complies with the requirements of 10 CFR 50.46 (b)(1)-(4).

2.4.1.2 <u>Post LOCA Long-Term Cooling</u>

The licensee's request stated that its analyses were performed for 1772 MWt rather than at the currently licensed value of 1650 MWt, but that it was not requesting an increase from 1650 MWt at this time. Consequently, the NRC staff limited its consideration to conditions consistent with operation at the currently licensed value. With respect to the currently licensed value, the reanalysis introduces a conservatism into the previously approved licensing basis analyses related to switchover from the injection phase to the recirculation phase, and for long-term cooling, that is sufficient to address the effect of the change in refueling water storage tank and accumulator boric acid concentration. Further, there are no concerns relative to a high heat flux within and from the fuel rods due to the relatively low heat generation rate associated with long term cooling. Therefore, the NRC staff determined that it was not necessary to review the licensee's reanalyses at this time. The NRC staff also finds that a review of the analyses will be required if the licensee requests an increase in thermal power from the currently licensed value of 1650 MWt. Similarly, the NRC staff further finds that the requirements for post-LOCA long term cooling are met for a thermal power level of 1650 MWt, because the conservatism introduced into the analysis of record via the assumption change is sufficient to balance the effect of the change in the boric acid concentrations.

2.4.2 <u>Non-LOCA Transients and Accidents</u>

The licensee reanalyzed the non-LOCA transient and accident analyses and evaluations to support implementation of the 422V+ fuel. These analyses were performed at a rated core power of 1772 MWt with other initial conditions consistent with this rated power level. For events where DNB is a primary concern or when revised thermal design procedure (RTDP) methodology is applied, a rated core power of 1772 MWt or a NSSS power of 1780 (including reactor coolant pump [RCP] pump heat of 8 MWt) are assumed in the analyses. For events primarily concerned with peak fuel temperatures or peak system pressures, or when STDP methodology is applied, 102 percent of rated core power of 1772 MWt or NSSS power of 1780 MWt are assumed in the analyses to account for instrument uncertainties. The initial conditions and major assumptions for the non-LOCA analyses are listed in Attachment B, Tables 5.1-2 through 5.1-7 of the licensee's submittal (Reference 2).

The licensee's analyses were performed using NRC-approved computer codes and methodologies including RETRAN, VIPRE, FACTRAN, TWINKLE, and LOFTRAN. In response to an NRC staff request (Reference 2, Attachment 3 RAI, Question 35), the licensee provided a tabulation of the computer codes used for each event analyzed, the approval status of each computer code, restrictions of each computer code and justification for using these codes for KNPP safety analyses. The NRC staff has reviewed the licensee's submittal and concludes that the licensee's analyses were performed using appropriate computer codes and methodologies.

The licensee's submittals (References 1 and 2) included discussions of the analyses and results for the proposed Westinghouse fuel type only. For the fuel transition, 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.

The licensee also provided justification that the safety analyses performed at a thermal power of 1772 MWt bound the plant operation at the current rated power level (Reference 2, Attachment 3 RAI, Question 38). The licensee based its conclusion on the following discussion. If rated power level is the only difference between analyses, the higher rated power analyses will by definition bound lower rated power analyses for analyses that are either full power limited or zero power limited. Since rated power level also affects at-power initial conditions assumed in the analyses, the licensee provided a tabulation of the assumed initial plant conditions for analyses at both current rated power and at 1772 MWt core power. It is indicated that all of the safety analysis input assumptions bound the operation of KNPP at both the current power and at 1772 MWt. Therefore, the NRC staff concludes that the safety analyses performed at 1772 MWt bound the plant operation at the current power level.

2.4.2.1 Uncontrolled RCCA Withdrawal From a Subcritical Condition

2.4.2.1.A Uncontrolled RCCA Withdrawal From a Subcritical Condition Regulatory Evaluation

An uncontrolled RCCA withdrawal from the 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 reactor protection system 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 protection system being designed 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.

2.4.2.1.B Uncontrolled RCCA Withdrawal From a Subcritical Condition Technical Evaluation

The licensee reanalyzed the uncontrolled RCCA bank withdrawal from subcritical event in support of the fuel upgrade. The licensee reanalyzed this event using a core power level of 1772 MWt. Although this event is initiated from a subcritical condition, the event relies on the power-range high neutron flux reactor trip (low setting) which is based on a percentage (%) of rated thermal power. The licensee used a conservative trip value of 35 percent in the analysis. The licensee employed conservative parameter values for this analysis and provided detailed information on the parameters used in Reference 2, Attachment B and Reference 2, Attachment 3 RAI, Question 30. The licensee verified that the values of all parameters used in the analyses remain bounding for each future core reload design, in accordance with current KNPP reload design methodology (Reference 22).

The licensee analyzed this event using the NRC-approved computer codes TWINKLE (Reference 23), FACTRAN (Reference 24) and Westinghouse VIPRE (Reference 18) and the methods described in the KNPP USAR. The spatial neutron kinetics code TWINKLE is used to calculate the core average nuclear power transient, including the various core feedback effects such as Doppler and moderator reactivity. FACTRAN uses the average nuclear power calculated by TWINKLE and performs fuel rod transient heat transfer calculations to determine the average heat flux and temperature transients. The peak core-average heat flux calculated by FACTRAN is used in VIPRE to calculate the transient DNBR values. The licensee provided information which demonstrates that all code conditions and restrictions are satisfied for the application of these codes to KNPP (Reference 2, Attachment 3 RAI, Question 35). The acceptance criteria for this event include DNBR remaining above the acceptance limit, and fuel temperature remaining below the melt temperature. The licensee provided results for this event which demonstrate that all acceptance criteria are satisfied (Reference 2, Attachment B and Reference 2, Attachment 3 RAI, Question 30). The minimum DNBR result of 1.588 is above the acceptance limit of 1.39 (STDP methodology and W-3 DNB correlation applied), and the peak fuel centerline temperature of 2685 °F is below the acceptance limit of 4746 °F.

2.4.2.1.C Uncontrolled RCCA Withdrawal From a Subcritical Condition Conclusion

The NRC staff finds that the licensee's analyses were performed using acceptable analytical models. The NRC staff has reviewed the licensee's analyses of the uncontrolled RCCA withdrawal from a subcritical condition and, for the reasons set forth above, 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. 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. Therefore, the NRC staff finds the proposed fuel upgrade acceptable with respect to the uncontrolled RCCA withdrawal from a subcritical condition.

2.4.2.2 Uncontrolled RCCA Withdrawal At Power

2.4.2.2.A Uncontrolled RCCA Withdrawal At Power Regulatory Evaluation

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 anticipated operational occurrence 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 reactor protection system 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 protection system being designed 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.

2.4.2.2.B Uncontrolled RCCA Withdrawal At Power Technical Evaluation

The licensee reanalyzed the uncontrolled RCCA bank withdrawal at power transient in support of the fuel upgrade. The licensee reanalyzed this event using a core power level of 1772 MWt. The event is analyzed using the RTDP methodology, therefore, initial reactor power, pressure and RCS temperatures are assumed to be at their nominal values. The licensee employed conservative parameter values for this analysis and provided detailed information on the parameters used in Reference 2, Attachment B. The analysis assumes a spectrum of possible reactivity insertion rates up to a maximum positive reactivity insertion rate greater than that occurring with the simultaneous withdrawal, at maximum speed of two sequential RCCA banks having the maximum differential rod worth. Power levels of 10 percent, 60 percent, and 100 percent of full power are analyzed. Depending on the initial power level and the rate of reactivity insertion, the reactor may be tripped and the RCCA withdrawal terminated by a number of reactor trip signals. The licensee's analyses credit both the power-range high neutron flux trip (118 percent of nominal full power) and the OTΔT reactor trip functions. The licensee verified that all values of parameters used in the analyses remain bounding for each future core reload design, in accordance with current KNPP reload design methodology (Reference 22). The NRC staff concludes that the licensee's analyses were performed using acceptable analytical models and appropriately conservative input parameter values.

The licensee analyzed the RCCA withdrawal at power transient using the NRC-approved RETRAN computer code (Reference 25) and the methods described in the KNPP USAR. The RETRAN computer code simulates plant systems, components and control systems, and computes transient parameters such as temperatures, pressures and power level. RETRAN can also be used to approximate transient DNBR values using the core thermal safety limit curves. The licensee provided justification for calculation of DNBR values using the RETRAN code in response to an NRC staff RAI (Reference 2, Attachment 3 RAI, Question 31.a). Additionally, the licensee provided information which demonstrates that all RETRAN code conditions and restrictions are satisfied for this application of RETRAN to KNPP (Reference 2, Attachment 3 RAI, Question 35).

The acceptance criteria for this event include DNBR remaining above the acceptance limit, and fuel temperature remaining below the melt temperature. The licensee provided results for this event which demonstrate that all acceptance criteria are satisfied (Reference 2, Attachment B and Reference 2, Attachment 3 RAI, Question 31.b). For the limiting cases, the minimum DNBR result of 1.46 is above the acceptance limit of 1.34 (RTDP methodology and WRB-1 DNB correlation applied), and the peak fuel centerline temperature remains below the acceptance limit. The fuel centerline temperature criteria is demonstrated by maintaining peak heat flux below the limiting value (Reference 2, Attachment 3 RAI, Question 31.b). The licensee also performed analyses and provided results which demonstrate that RCS and main steam system pressures remain below 110 percent of the corresponding design limits. Because the licensee's analyses were performed using acceptable analytical models and appropriate plant-specific inputs, the staff finds the results are acceptable.

2.4.2.2.C Uncontrolled RCCA Withdrawal At Power Regulatory Evaluation Conclusion

The NRC staff concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff has reviewed the licensee's analyses of the uncontrolled RCCA withdrawal at power and, for the reasons set forth above, 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. 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. Therefore, the NRC staff finds the proposed fuel upgrade acceptable with respect to the uncontrolled RCCA withdrawal at power.

2.4.2.3 RCCA Misalignment

2.4.2.3.A RCCA Misalignment Regulatory Evaluation

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 reactor protection system being designed to automatically initiate appropriate systems to ensure that are not exceeded as a result of AOOs; and (3) GDC-25 for the protection system being designed 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.

2.4.2.3.B RCCA Misalignment Technical Evaluation

The RCCA misalignment transient analyses consider dropped full-length RCCA's, dropped full-length RCCA banks, and statically misaligned full-length RCCA's. The licensee reanalyzed the RCCA misalignment events in support of the fuel upgrade. The licensee reanalyzed these events using a core power level of 1772 MWt. The events are analyzed using the RTDP methodology, therefore, initial reactor power, pressure and RCS temperatures are assumed to be at their nominal values. The licensee performed the analyses in accordance with the NRC-approved methodology described in WCAP-11394-P-A (Reference 26) and performed the comparisons identified as a condition of its use (Reference 2, Attachment 3 RAI, Question 35). This methodology involves analyses to determine the most limiting statepoints, the limiting radial power peaking factor and the DNB at the limiting conditions. The acceptance criteria for this event include DNBR remaining above the acceptance limit, and fuel temperature remaining below the melt temperature. The methodology ensures that the DNBR acceptance criteria are satisfied by comparing the transient $F^N_{\Delta H}$ values to the $F^N_{\Delta H}$ limits. The fuel centerline melt criteria is evaluated based on the maximum transient F_Q value.

The licensee performed these analyses using the LOFTRAN (Reference 27), VIPRE (Reference 18) and ANC (References 14 and 15) computer codes. The LOFTRAN computer code simulates plant systems, components and control systems, and computes transient parameters such as temperatures, pressures and power level (transient RCS statepoints). The VIPRE code is used to demonstrate that the DNBR acceptance criteria are satisfied. The ANC code is used to calculate steady state power distributions. The licensee provided information which demonstrates that all code conditions and restrictions are satisfied for this application of these codes to KNPP (Reference 2, Attachment 3 RAI, Question 35). The licensee provided results for this event which demonstrate that all acceptance criteria are satisfied (Reference 2, Attachment B and Reference 2, Attachment 3 RAI, Question 32). In all cases, the transient $F^N_{\Delta H}$ values demonstrate that the DNBR acceptance criteria are satisfied, and the fuel temperature remains below the melt temperature. The licensee verified that the values of all parameters used in the analyses remain bounding for each future core reload design, in accordance with current KNPP reload design methodology (Reference 22).

2.4.2.3.C RCCA Misalignment Conclusion

The NRC staff concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff has reviewed the licensee's analyses of the RCCA misalignment transients and, for the reasons set forth above, 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. 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. Therefore, the NRC staff finds the proposed fuel upgrade acceptable with respect to the RCCA misalignment transients.

2.4.2.4 Chemical and Volume Control System Malfunction

2.4.2.4.A Chemical and Volume Control System Malfunction Regulatory Evaluation

Unborated water can be added to the RCS 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.

2.4.2.4.B Chemical and Volume Control System Malfunction Technical Evaluation

The licensee reanalyzed the CVCS malfunction event in support of the proposed fuel upgrade. Boron dilution events during refueling, startup and power operation were reanalyzed. The method of analysis employed is consistent with that currently described in the KNPP USAR and was performed using conditions at a core power level of 1772 MWt.

The analyses are performed to determine the amount of time available for operator action prior to loss of shutdown margin. As long as the available operator action times are sufficient to prevent loss of shutdown margin, it can be concluded that fuel cladding damage and RCS overpressurization acceptance criteria will also be satisfied. The SRP specifies a minimum of 15 minutes between time of indication (alarm) and time for loss of shutdown margin for the at-power and startup conditions, and 30 minutes for the refueling condition. The licensee provided results for these analyses which demonstrate that the operator action time acceptance criteria are satisfied for boron dilution events during refueling, startup and power operations (Reference 2, Attachment B and Reference 2, Attachment 3 RAI, Question 33). The licensee noted in the RAI response (Reference 2, Attachment 3 RAI, Question 33) that the current KNPP licensing basis does not require specific criteria for operator action times for the at-power event, but committed to add this criterion to the KNPP licensing basis beginning with the next operating Cycle (Cycle 26).

The licensee's USAR states that the maximum dilution flow rate (80 gpm) occurs with two charging pumps operating and three letdown orifices in service. However, the licensee's USAR analysis of record assumed a larger, conservative flow rate of 180 gpm, corresponding to all three charging pumps operating at full flow (60 gpm/pump). The licensee's initial reanalysis for the fuel upgrade (Reference 1) for this event assumed a maximum dilution flow rate of 80 gpm. The NRC staff questioned this reduction in conservatism (Reference 2, Attachment 3 RAI, Question 33) and the licensee responded by revising the analyses to include a maximum dilution flow rate of 120 gpm. The normal operating state at KNPP is to have two charging pumps operating, one in manual and one in automatic control, and one pump not operating. An initiating event is assumed that results in a maximum of two pumps delivering their maximum flow for a total dilution rate of 120 gpm. The licensee provided justification for this assumption by stating that operation with more than two charging pumps operating is precluded by procedural administrative control and operator training. Additionally, the charging pumps do not have an automatic start feature such that an additional failure would result in the starting of the third pump. Based on this, the NRC staff finds the maximum dilution flow of 120 gpm assumed in the licensee's analyses to be acceptable.

2.4.2.4.C Chemical and Volume Control System Malfunction Conclusion

The NRC staff concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff has reviewed the licensee's analyses of the CVCS malfunction transient and, for the reasons set forth above, 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. 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. Therefore, the NRC staff finds the proposed fuel upgrade acceptable with respect to the CVCS malfunction transient.

2.4.2.5 Startup of an Inactive Reactor Coolant Loop

2.4.2.5.A Startup of an Inactive Reactor Coolant Loop Regulatory Evaluation

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 or damaged 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

2.4.2.5.B Startup of an Inactive Reactor Coolant Loop Technical Evaluation and Conclusion

The KNPP TS limits the reactor power to <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 cold-leg 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. The NRC staff agrees with the 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.

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

2.4.2.6.A Excessive Heat Removal Due to Feedwater Temperature Reduction or Flow Increase Regulatory Evaluation

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 covered 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 reactor protection system, 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 20 as it relates the reactor protection system 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.

2.4.2.6.B Excessive Heat Removal Due to Feedwater Temperature Reduction or Flow Increase Technical Evaluation and Conclusion

The licensee reanalyzed this event using the RETRAN code. The safety analysis minimum DNBR limit for this event is 1.34. The licensee calculated minimum DNBR for limiting case is 1.709 for hot full power case and 2.837 for hot zero power case. The NRC staff has reviewed the licensee's analyses of the excessive heat removal events due to feedwater system malfunction and finds that the licensee has used the NRC-approved RETRAN code within the range of applicability for which it was approved and has used appropriate plant-specific inputs. Accordingly, the NRC staff concludes that the licensee's analyses have correctly and conservatively accounted for operation of the plant at the initial nuclear steam supply system (NSSS) power level of 1780 MWt (maximum core power of 1772 MWt plus RCP heat of 8 MWt) and that the 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 SAFDLs and the RCPB pressure limits will not be exceeded as a result of these events. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDC 10, 15, 20, and 26. Therefore, the staff finds the licensee's analysis of the excessive heat removal events is acceptable to support the use of 422V + fuel.

2.4.2.7 Excessive Load Increase Incident

2.4.2.7.A Excessive Load Increase Incident Regulatory Evaluation

The regulatory requirements for this analysis are the same as for the excess heat removal due to feedwater system malfunction event discussed above.

2.4.2.7.B Excessive Load Increase Incident Technical Evaluation and Conclusion

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 steam generator steam flow that cause a power mismatch between the reactor core power and the steam generator 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 reactor protection system. 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 (set here at 1772 MWt) 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 results of the licensee's evaluation for this event showed that the minimum DNBR remains above the safety analysis limit value. Since this is a cooldown transient, peak primary and secondary system pressures are not challenged. The NRC staff finds the results of the licensee's evaluation acceptable because the acceptance criteria for this event are satisfied. Therefore, the NRC staff finds the licensee's analysis of the excessive load increase event is acceptable to support the use of 422V + fuel.

2.4.2.8 Loss of Reactor Coolant Flow

2.4.2.8.A Loss of Reactor Coolant Flow Regulatory Evaluation

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 SAFDLs 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 reactor protection system, 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 auxiliary systems 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 structures, systems, and components 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.

2.4.2.8.B Loss of Reactor Coolant Flow Technical Evaluation

The licensee reanalyzed this event using the RETRAN and VIPRE computer codes. The licensee used these NRC-approved codes in conformance with their intended ranges of applicability and with appropriate plant specific inputs. A maximum steam generator tube plugging level of 10 percent was assumed in the analysis. After the most limiting case of a complete loss of flow event, reactor trip is initiated on the RCS low-flow signal. The results of the licensee's analysis shows that the calculated peak RCS pressure is 2350 psia which is less than the acceptance criteria of the design plus 10 percent (2750 psia) and the KNPP TS 2.2, "Safety Limit - Reactor System Pressure," limit of 2750 psia. The licensee stated that the peak secondary system pressure for this event is bounded by that for the loss of load event, and is below the acceptance limit of 1210 psia, which is the secondary system design pressure plus 10 percent. The licensee calculated minimum DNBR for the limiting case of a complete loss of reactor coolant flow to be 1.386, which is above the safety analysis minimum DNBR limit of 1.34.

2.4.2.8.C Loss of Reactor Coolant Flow Conclusion

The NRC staff has reviewed the licensee’s analyses of the decrease in reactor coolant flow event and, for the reasons set forth above, concludes that the licensee’s analyses were performed using acceptable analytical models and that the analyses have correctly and conservatively accounted for operation of the plant with 422V+ fuel at a core power level of 1772 MWt. 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 this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDC 10, 15, 17, and 26. Therefore, the NRC staff finds the licensee’s analysis of the loss of coolant flow events is acceptable to support the use of 422V + fuel.

2.4.2.9 Locked Rotor Accident

2.4.2.9.A Locked Rotor Accident Regulatory Evaluation

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 reactor protection system, 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 structures, systems, and components important to safety will function; GDC 27 and GDC 28 as they relate to the RCS being designed 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 rapidly 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 for the maximum clad temperature of 2700 ^{o}F to assure that a coolable core geometry is maintained.

2.4.2.9.B Locked Rotor Accident Technical Evaluation

The licensee reanalyzed this event using the RETRAN, VIPRE and FACTRAN Computer Codes. The licensee used these NRC-approved codes in conformance with their intended ranges of applicability and with appropriate plant specific inputs. A maximum steam generator tube plugging level of 10 percent was assumed in the analysis. Following the locked rotor, reactor trip is initiated on the RCS low-flow signal. A loss of off-site power is assumed at the time of reactor trip. The results of the licensee's analysis shows that the peak clad temperature is 1900 ^{o}F which is less than the licensing basis criterion of 2700 ^{o}F. The calculated peak RCS pressure is 2683 psia, which is less than the acceptance criterion of 2750 psia, which is the design pressure plus 10 percent. The licensee stated that the peak secondary system pressure for this event is bounded by that for the loss of load event, and is below the acceptance limit of 1210 psia, which is the secondary system design pressure plus 10 percent. The total percentage of fuel rods calculated to experience DNB is less than licensing basis criterion of 50 percent which assures that the radiological consequences are within the 10 CFR Part 100 guidelines. The NRC staff previously evaluated dose assessment in its radiological consequence analyses for the postulated design-basis accidents in Kewaunee License Amendment 166, dated March 17, 2003, and the NRC staff determined that the changes are acceptable for meeting the dose acceptance criteria specified in 10 CFR 50.67.

2.4.2.9.C Locked Rotor Accident Conclusion

The NRC staff has reviewed the licensee's analyses of the sudden decrease in core coolant flow events and, for the reasons set forth above, concludes that the licensee's analyses were performed at 1772 MWt using acceptable analytical models and that the analyses have correctly and conservatively accounted for operation of the plant with 422 V+ fuel. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the RCPB pressure limits will not be exceeded, the RCPB will behave in a nonbrittle manner, the probability of propagating fracture of the RCPB is minimized, and abundant core cooling will be provided.   Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDC 17, 27, 28, and 31. Therefore, the NRC staff finds the licensee's analysis of the sudden decrease in core coolant flow events is acceptable to support the use of 422V + fuel.

2.4.2.10 Loss of External Electric Load

2.4.2.10.A Loss of External Electric Load Regulatory Evaluation

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 auxiliary systems 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 structures, systems, and components 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.

2.4.2.10.B Loss of External Electric Load Technical Evaluation

The licensee reanalyzed this event using the RETRAN computer Code. The licensee used this NRC-approved code in conformance with its intended range of applicability and with appropriate plant specific inputs. A maximum steam generator tube plugging level of 10 percent was assumed in the analysis. After the loss of the external electric load, reactor trip is initiated by various reactor protection system. The results of the licensee’s analysis shows that the calculated peak RCS pressure is 2697 psia, which is less than the KNPP TS limit of 2750 psia. The peak secondary system pressure of this event is 1202 psia which is below its limit of 1210 psia. The calculated minimum DNBR is 1.74 which is above the minimum DNBR limit of 1.34 for this event.

2.4.2.10.C Loss of External Electric Load Conclusion

The NRC staff has reviewed the licensee’s analyses of decreases in heat removal by the secondary system and, for the reasons set forth above, concludes and that the analyses were performed at a thermal power of 1772 MWt using acceptable analytical models and that the licensee’s analyses have correctly and conservatively accounted for operation of the plant with 422V+ fuel. 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 these events. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDC 10, 15, 17, and 26. Therefore, the NRC staff finds the licensee’s analysis of the loss of the external electric load event is acceptable to support the use of 422V + fuel.

2.4.2.11 Loss of Normal Feedwater

By its letter dated February 27, 2003 (Reference 2), the licensee has indicated that the loss of normal feedwater analysis has been retracted. Since the loss of normal feedwater transient is driven by decay heat, and the decay heat model is independent of fuel design, the NRC staff concluded that this analysis is not required to support the use of 422V + Fuel at KNPP.

2.4.2.12 Anticipated Transients Without Scram

An anticipated transient without scram (ATWS) mitigation system actuation circuitry (AMSAC) has been installed at KNPP, and therefore, the requirements of 10 CFR 50.62(b) have been satisfied. After implementation of fuel transition, the AMSAC will continue to be operable at KNPP in compliance with the requirements of the ATWS rule. As a supplement to AMSAC, a diverse scram system has been installed at KNPP. The licensee meets the requirements of 10 CFR 50.62(b); therefore, no additional ATWS analyses are required to support the use of 422V + Fuel at KNPP.

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

2.4.2.13.A Loss of AC Power to the Plant Auxiliaries Regulatory Evaluation

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 reactor coolant circulation pumps. 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 SAFDLs not exceeded during normal operation including AOOs; GDC 15 as it relates to the RCS and its associated auxiliary systems 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 that SAFDLs are not exceeded during normal operation, including AOOs. Specific review criteria are contained in SRP Section 15.2.6.

2.4.2.13.B Loss of AC Power to the Plant Auxiliaries Technical Evaluation

The licensee reanalyzed this event using the RETRAN computer Code. The licensee used this NRC-approved code in conformance with its intended range of applicability and with appropriate plant specific inputs. The analysis assumed an initial power level of 102 percent of NSSS power of 1780 MWt. Following the loss of ac power to the plant auxiliaries, a reactor trip is initiated on steam generator low-low level signal. To maximize the potential for pressurizer filling, the power operated relief valves (PORVs) are assumed to be operable. The peak pressures for primary and secondary systems are not calculated in this analyses since they are bounded by the results of the loss of external electrical load event discussed above. The results of the licensee's analysis shows that the calculated peak pressurizer water volume is 698 cubic feet, which is less than the accident analysis limit of 1,010 cubic feet. Therefore, the pressurizer will not become water solid and no liquid will be discharged through the PORV and/or safety valves. Because this is a heat up transient, DNBR will not be challenged.

2.4.2.13.C Loss of AC Power to the Plant Auxiliaries Conclusion

The NRC staff has reviewed the licensee's analyses of the loss of nonemergency ac power to station auxiliaries event and, for the reasons set forth above, concludes that the licensee's analyses were performed at a thermal power of 1772 MWt using acceptable analytical models and that the analyses have correctly and conservatively accounted for operation of the plant using 422V+ fuel. 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 this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDC 10, 15, and 26.

Therefore, the NRC staff finds the licensee's analysis of the loss of the ac power to the plant auxiliaries event is acceptable to support the use of 422V + fuel.

2.4.2.14 Steamline Break

2.4.2.14.A Steamline Break Regulatory Evaluation

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 reactor protection system; 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 important to safety; GDC 27 and GDC 28 as they relate to reactivity control systems being designed 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 rapidly propagating fracture is minimized; and GDC 35 as it relates to the requirement to provide a system providing abundant emergency core cooling. Specific review criteria are contained in SRP Section 15.1.5.

2.4.2.14.B Steamline Break Technical Evaluation

The licensee reanalyzed this event using the RETRAN, VIPRE, ANC and PHOENIX-P computer Codes. The licensee used these NRC-approved codes in conformance with their intended ranges of applicability and with appropriate plant specific inputs. Following a steamline break, a reactor trip is initiated by a safety injection signal or overpower reactor trip signals. A maximum steam generator tube plugging level of 10 percent was assumed in the analysis in order to maximize the cooldown of the RCS. The analysis also assumes the end of life shutdown margin and the most reactive RCCA stuck in its fully withdrawn position. The event is analyzed with and without off-site power available. The main feedwater and main steamlines will be isolated on the safety injection signal following a steamline break. The results of the licensee's analysis shows that the limiting case for core response is the event with off-site power available. The calculated minimum DNBR is 2.29 which is above the allowable minimum DNBR limit of 1.472 for this event. Therefore, there is no fuel failure in this event. The primary and secondary pressures will be decreasing from their initial conditions. Therefore, the peak pressures are not a concern in this event.

2.4.2.14.C Steamline Break Conclusion

The NRC staff has reviewed the licensee's analyses of steam system piping failure events and, for the reasons set forth above, concludes that the licensee's analyses were performed at a thermal power of 1772 MWt using acceptable analytical models and that the analyses have correctly and conservatively accounted for operation of the plant using 422V+ fuel. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the RCPB pressure limits will not be exceeded, the RCPB will behave in a nonbrittle manner, the probability of rapidly propagating fracture of the RCPB is minimized, and abundant core cooling will be provided. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDC 17, 27, 28, 31, and 35. Therefore, the NRC staff finds the licensee's analysis of the steamline break accident is acceptable to support the use of 422V + fuel.

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

2.4.2.15.A Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection) Regulatory Evaluation

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:

a. Reactivity excursions should not result in a radially averaged enthalpy greater than 280 cal/gm at any axial location in any fuel rod.

b. 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 Boiler and Pressure Vessel Code.

2.4.2.15.B Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection) Technical Evaluation

The licensee's analyses employ a more restrictive acceptance criteria of 200 cal/gm for maximum average fuel pellet enthalpy at the hot spot, and also provide that fuel melting be limited to the innermost 10 percent of the fuel pellet at the hot spot, independent of the pellet enthalpy limit. These criteria are consistent with the approved Westinghouse reload safety evaluation methodology (Reference 13) and the Westinghouse RCCA Ejection analysis methodology described in WCAP-7588, Revision 1-A (Reference 28).

The licensee reanalyzed the RCCA ejection accident in support of the fuel upgrade at a power level of 1772 MWt. Uncertainties in core power and other input parameters were included because the STDP methodology was applied. As such, the full power cases were analyzed at 102 percent of this power level. Consistent with current methodology, RCCA ejection accidents were analyzed at hot full power (HFP) and hot zero power (HZP) conditions, for both beginning and end of life cores. The licensee employed conservative parameter values for this analysis and provided detailed information on the parameters used in Reference 2, Attachment B. The licensee verified that all parameters used in the analyses remain bounding for each future core reload design, in accordance with current KNPP reload design methodology (Reference 22).

The licensee reanalyzed the RCCA Ejection accident in accordance with NRC-approved methodology described in WCAP-7588, Revision 1-A (Reference 28) and provided information which demonstrates that all methodology conditions and restrictions are satisfied for the application of this method to KNPP (Reference 2, Attachment 3 RAI, Question 35). The calculation includes a neutron kinetic analysis and a hot-spot fuel heat transfer analysis. The licensee performed these calculations using the NRC-approved computer codes TWINKLE (Reference 23) and FACTRAN (Reference 24). The spatial neutron kinetics code TWINKLE is used to calculate the core nuclear power transient, including the various core feedback effects such as Doppler and moderator reactivity. FACTRAN uses the nuclear power calculated by TWINKLE and performs fuel rod transient heat transfer calculations to determine the fuel enthalpy and temperature transients. The licensee provided information which demonstrates that all code conditions and restrictions are satisfied for the application of these codes to KNPP (Reference 2, Attachment 3 RAI, Question 35).

The licensee provided results for this event which demonstrate that all acceptance criteria are satisfied (Reference 1 and Reference 2, Attachment B). For all cases, the maximum fuel pellet enthalpy remained below 200 cal/gm. For the HFP cases, the peak hot-spot fuel centerline temperature reached the fuel melting temperature; however, melting was restricted to less than 10 percent of the pellet. For the HZP cases, no fuel melting was predicted to occur. The licensee credits the generic Westinghouse analyses for calculations of the RCS pressure and number of rods in DNB (Reference 28). This is consistent with the licensee's current methodology as described in the KNPP USAR. The licensee verified that this generic case continues to bound the worst-case KNPP RCCA Ejection accident considering the conditions representative of the fuel upgrade (Reference 35). The results of this generic analysis demonstrated that the RCS pressure acceptance criteria are satisfied. The number of rods in DNB is used in the radiological consequences evaluation, for which KNPP currently assumes 15 percent of rods in DNB for the RCCA Ejection accident. The generic Westinghouse analysis (Reference 28) demonstrates an upper limit on number of rods in DNB of 10 percent, which is well within the 15 percent value assumed in the radiological analysis. The licensee concluded that the 10 percent generic analysis value remains valid for the fuel upgrade. 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, because they were performed using acceptable analytical models and appropriate plant-specific inputs.

2.4.2.15.C Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection) Conclusion

The NRC staff has reviewed the licensee's analyses of the rod ejection accident and, for the reasons set forth above, concludes that the licensee's analyses were performed at a thermal power of 1772 MWt using acceptable analytical models and that the analyses have adequately accounted for operation of the plant for the fuel upgrade. 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. Therefore, the NRC staff finds the proposed fuel upgrade acceptable with respect to the rod ejection accident.

2.5 <u>RTDP Uncertainty Calculations</u>

2.5.A RTDP Uncertainty Calculations Regulatory Evaluation

WCAP-11397-P-A, “Revised Thermal Design Procedure,” provides guidance for using a statistical methodology to calculate departure from nucleate boiling limits. The guidance provided in WCAP-11397-P-A was used by the NRC staff to assess the instrument uncertainty calculations documented in WCAP-15591.

GL 88-16 provides licensee’s guidance for relocating cycle-dependent variables from TS, provided that the values of these variables are included in a COLR, are determined with NRC-approved methodologies referenced in the TS, and are reported to the NRC as they are made. Relocating cycle-dependent variables to the COLR reduces the regulatory burden on the licensee by allowing changes to these variables using 10 CFR 50.59, “Changes, tests, and experiments,” as opposed to changing the TS using 10 CFR 50.90, “Application for amendment of license or construction permit.” Cycle-specific Westinghouse plant parameters approved by GL 88-16 for relocation to the COLR include: (1) moderator temperature coefficient, (2) shutdown bank insertion limits, (3) control bank insertion limits, (4) axial flux difference limits, (5) nuclear heat flux hot channel factor limit (F_Q), (6) nuclear enthalpy rise hot channel factor limit ($F^N_{\Delta H}$), (7) refueling boron concentration limit, and (8) shutdown margin.

WCAP-14483-A extended the scope of GL 88-16 to provide licensee’s guidance for relocating from the TS to the COLR reactor core safety limit curves (but not the safety limits [SLs] themselves); cycle-specific DNB parameter limits, with the addition of the DNBR design limit (but not DNBR SLs); the fuel centerline melt temperature limit; OTΔT and OPΔT parameter constants, including f(ΔI) slope and breakpoint values; and the minimum reactor coolant flow design limit.

The NRC staff finds acceptable the regulatory requirements and guidance identified by the licensee. These regulatory requirements and supporting guidance were used by the NRC staff in its evaluation of the instrumentation uncertainties reported for the licensee by Westinghouse in WCAP-15591; and in its evaluation of the licensee’s proposed relocation of OTΔT and OPΔT parameter constants including f(ΔI) slope and breakpoint values from the TS to the COLR.

2.5.B Instrument Uncertainty Methodology Technical Evaluation

As stated in WCAP-15591, four operating parameter uncertainties are used in the uncertainty analysis of the RTDP. These parameters are pressurizer pressure, primary coolant temperature (T_{avg}), reactor power, and RCS flow. These parameters are frequently monitored by the licensee, and several of the parameters are used for control purposes. Reactor power is monitored by performing a secondary side heat balance (power calorimetric measurement) at least once every 24 hours. RCS flow is monitored by performing a calorimetric RCS flow measurement at the beginning of each cycle. The RCS cold leg loop flow indicators are evaluated using the results of the calorimetric RCS flow measurement. Pressurizer pressure is a controlled parameter that is measured with pressure instrumentation. T_{avg} is a controlled parameter via the temperature input to the rod control system. The uncertainties in pressure and T_{avg} measurements are affected by the characteristics of their control and indication systems.

The RTDP methodology is used to calculate plant DNBR design limits by statistically combining plant operating parameter uncertainties. The RTDP methodology provides that variations in plant operating parameters be justified. The purpose of WCAP-15591 was to document the determination of pressure, temperature, power, and RCS flow uncertainties that are applicable to KNPP for power levels up to 1757 MWt NSSS power when using feedwater venturis for 18-month fuel cycles +25 percent, in accordance with the plant TS, and for a full power T_{avg} window from 556.3 °F to 573.0 °F.

2.5.C Instrument Uncertainty Methodology Conclusion

The NRC staff reviewed the RTDP methodology described in WCAP-15591, using as a reference, the NRC staff safety evaluation of the RTDP methodology, which is documented in the NRC-approved WCAP-11397-P-A, dated January 17, 1989, and licensee responses to NRC staff questions, as documented in a letter to the NRC dated February 27, 2003. This methodology may be applied to 422V+ fuel analyses at KNPP because the conditions at KNPP in which the methodology would be applied are bounded by the conditions set forth in WCAP-11397-P-A and for which the NRC staff approved the use of the methodology, and the licensee has otherwise satisfied the conditions for use set forth in the NRC staff's safety evaluation approving WCAP-11397-P-A. On the basis of its review, the NRC staff concludes that the RTDP methodology used in WCAP-15591 is the same methodology as the methodology described in WCAP-11397-P-A, with the plant-specific inputs within the intended ranges of the approved methodology.

The NRC staff reviewed the control system and indication uncertainties listed in WCAP-15591, and the process by which these uncertainties were combined. The licensee stated that the uncertainty terms listed in WCAP-15591 were obtained from manufacturer's specifications; calibration tolerances for the KNPP instrumentation are based on KNPP calibration procedures; and the magnitude of instrument drift allowances are equivalent to plant procedure calibration tolerances. The NRC staff finds this response acceptable for defining the source of the values listed in WCAP-15591.

The NRC staff performed confirmatory calculations using the data listed in WCAP-15591 to verify the calculated results of the WCAP-15591 statistical-based analyses. The calculations performed by the NRC staff resulted in the same values calculated by Westinghouse for the licensee. The NRC staff, therefore, concludes that the RTDP methodology, as described in WCAP-11397-P-A, was correctly implemented in WCAP-15591.

On the basis of its review and confirmatory analyses, the NRC staff concludes that WCAP-15591 provides an acceptable basis for the uncertainty values used in associated licensing bases analyses for KNPP.

2.6 Dose Assessment

2.6.A Dose Assessment Regulatory Evaluation

The NRC staff utilized the regulatory guidance and safety evaluation provided in the following documents in performing this review:

1. NUREG-1431, Revision 2, “Standard Technical Specifications - Westinghouse Plants,”

2. Regulatory Issue Summary (RIS) 01-019, ”Deficiencies in the Documentation of Design Basis Radiological Analyses Submitted in Conjunction with License Amendment Requests,”

3. 10 CFR 50.67, and

4. Kewaunee License Amendment 166 dated March 17, 2003, concerning revision to the design-basis radiological analysis accident source term implementing the alternative source term.

2.6.B Dose Assessment Technical Evaluation and Conclusion

The NRC staff previously evaluated dose assessment in its radiological consequence analyses for the postulated design-basis accidents in Kewaunee License Amendment 166, dated March 17, 2003, and the NRC staff determined that the changes are acceptable for meeting the dose acceptance criteria specified in 10 CFR 50.67. There are no changes with respect to dose assessment from the use of 422V+ fuel, and therefore, the results of the NRC staff’s March 17, 2003, safety evaluation relating to Amendment 166 remain unchanged.

3.0 CHANGES OF TSs

3.1 TS 2.1.b - Safety Limits, Reactor Core

10 CFR Part 50, Appendix A, GDC 10 requires the reactor core and associated coolant, control, and protection systems be designed to assure that SAFDLs are not exceeded during steady state operation, normal operational transients, 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 reactor core safety limit curves, which now reside in the COLR, show the loci of points of thermal power, RCS pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting temperature, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the exit quality is within the limits defined by the DNBR correlation. Those DNB correlations used to generate or validate the safety limit curves should be included in TS 2.1.

The licensee proposes to add the Westinghouse WRB-1 DNB correlation and its correlation limit of 1.17 to KNPP TS 2.1. The WRB-1 DNB correlation is based entirely on rod bundle experimental data. The NRC staff has previously approved a 95/95 correlation limit (the probability that DNB will not occur on the most limiting fuel rod is at least 95 percent, at a 95 percent confidence level) of 1.17 for both Westinghouse 14x14 Optimized Fuel Assembly (OFA) and 15x15 OFA fuel designs (Reference 2, Attachment 3 RAI, Question 21). The WRB-1 DNB correlation and an associated correlation limit of 1.17 has also been shown to be acceptable for use with the Westinghouse VIPRE code (Reference 18), and has previously been approved for the Westinghouse 14x14 422V+ fuel design at Point Beach (Reference 19). The licensee demonstrated that the new mid-grid meets all design criteria of existing tested mid-grids that form the basis of the WRB-1 correlation database and that the WRB-1 correlation with a 95/95 correlation limit of 1.17 applies to the new mid-grid (Reference 21). Accordingly, the NRC staff finds that the WRB-1 correlation and a correlation limit of 1.17 is applicable for the proposed Westinghouse 14x14 422V+ fuel design to be used at KNPP.

The licensee uses the Framatome/ANP high thermal performance (HTP) and Westinghouse WRB-1 DNB correlations to generate and validate the safety limit curves, and therefore, both of these correlations and their respective correlation limits should be included in the KNPP TS 2.1 (Reference 2, Attachment 2 RAI, Question 1). The HTP DNB correlation applies to the current KNPP Framatome/ANP fuel with HTP spacers. The WRB-1 correlation applies to the Westinghouse 422V+ fuel. As set forth above, the NRC staff finds the application of the WRB-1 correlation and a correlation limit of 1.17 to be acceptable for KNPP for the fuel upgrade (Section 2.3 - Thermal and Hydraulic Design Evaluation). Based on this, the NRC staff finds that the proposed change to KNPP TS 2.1 is acceptable.

3.2 TS 2.3.a.3.A and TS 2.3.a.3.B - Overtemperature ΔT and Overpower ΔT Setpoints

10 CFR Part 50, Appendix A, GDC 10 requires the reactor core and associated coolant, control, and protection systems be designed to assure that SAFDLs are not exceeded during steady state operation, normal operational transients, and AOO’s. 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 Reactor Protection System, 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 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 parameter values. All constant and parameter values are currently specified in the KNPP COLR. KNPP employs NRC-approved methodology outlined in WCAP-8745-P-A (Reference 29) to calculate these parameter values. This staff approved methodology is currently referenced in KNPP TS 6.9. In accordance with NRC GL 88-16 (Reference 30), the licensee can revise these values using a staff approved methodology, without prior NRC review and approval. The licensee is proposing to change equation constant and parameter values as listed in Reference 2, Attachment B, Table 5.1-4, and proposes to change the Overpower ΔT trip setpoint f(ΔI) value to 0 for all ΔI. Additionally, the licensee proposes to revise the parameter value for T' from = to ≤ the value specified in the COLR, consistent with the Westinghouse Standard Technical Specifications (STS) (Reference 11). Because NRC-approved methodology is employed to calculate these quantities using appropriate cycle-specific input data, and the results of the licensee's safety analyses are acceptable, the NRC staff finds these changes to TS 2.3.a.3.A and TS 2.3.a.3.B to be acceptable.

3.3 <u>TS 3.10.b - Control Rod and Power Distribution Limits</u>

The licensee proposes a number of changes to KNPP TS 3.10.b involving power distribution limits. The changes are based on the licensee's core axial offset control methodology conversion from constant axial offset control (CAOC) to relaxed axial offset control (RAOC). The licensee plans to incorporate the following STS (Reference 11) sections using the RAOC specifications:

- 3.2.1.B, "Heat Flux Hot Channel Factor ($F_Q(z)$ (RAOC-W(z) Methodology)."
- 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)."
- 3.2.3.B, "Axial Flux Difference (AFD) (Relaxed Axial Offset Control (RAOC) Methodology."

The NRC staff previously approved the use of the Westinghouse developed RAOC methodology for performing power distribution control in Westinghouse-type PWRs in WCAP-10216-P-A (Reference 31). Application of the RAOC and F_Q methodologies requires changes to the AFD and F_Q TS listed above. Additionally, a reference to the NRC-approved RAOC methodology (Reference 31) must be added to the COLR administrative section of the KNPP TS. The licensee provided the basis and justification for applying the RAOC methodology in response to an NRC staff RAI (Reference 2, Attachment 2 RAI, Question 3).

The NRC staff reviewed the licensee's proposed TS changes (as marked-up in Reference 2, Attachment E), including limiting condition for operations, required actions, completion times, and surveillance requirements and frequencies, and found that they are consistent with STS (Reference 11) and are in conformance with the conditions for applicability at KNPP. The licensee requested an additional change to remove the requirement to verify AFD within limits if the associated alarms are out-of-service. The NRC staff approved relocation of this requirement from Westinghouse STS to plant administrative practices in Technical Specification Task Force (TSTF) 110, Revision 2 (Reference 32). As part of the removal of this requirement, the licensee will add the surveillance requirement contained in TSTF 110 Revision 2 to verify AFD within limits weekly (Reference 2, Attachment E, Table TS 4.1-1). The NRC staff finds that implementing this change is acceptable for KNPP because no safety functions are adversely effected by this change. The NRC staff finds the proposed KNPP TS changes associated with a conversion to RAOC methodology to be acceptable.

3.4 <u>TS 3.10.m (RCS Flow)</u>

The current TS 3.10.m regarding reactor coolant flow states that during steady state power operation, reactor coolant flow rate shall be >93,000 gallons per minute average per loop and greater than or equal to the limit specified in the COLR. Also, compliance with this flow requirement shall be demonstrated by verifying the reactor coolant flow during initial power escalation following each refueling between 70 percent and 95 percent power with plant parameters as constant as practical. The proposed change to this TS will require that reactor coolant total flow rate shall be > 178,000 gallons per minute. This required reactor coolant flow rate is consistent with the assumption used in the licensee's new safety analyses as is stated in the data provided in its response to the staff question regarding the initial conditions assumed in the safety analyses (Reference 2, Attachment 3 RAI, Question 36). The proposed change will also require the verification of this flow rate during the initial power escalation following each refueling at or above 90 percent power. The NRC staff has reviewed the licensee proposed changes and find that they are acceptable because the changes are consistent with the value assumed in the licensee's new analyses, and the required power level of 90 percent for the test is within the power range specified in the current TS.

3.5 TS 5.3 - Reactor Core

The licensee proposes to add wording to allow loading of fuel assemblies with zircaloy or ZIRLO cladding. The current Framatome/ANP fuel contains zircaloy cladding. The addition of an option for ZIRLO cladding addresses the Westinghouse 422V+ fuel. The NRC staff finds this change to be acceptable based on the acceptability of the Westinghouse 422V+ fuel design. The NRC staff's conclusion is supported by the discussions included throughout this safety evaluation.

3.6 TS 6.9 - Changes Associated with COLR References

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to state the TSs to be included as part of the license. The Commission's regulatory requirements regarding the content of plant TSs are set forth in 10 CFR 50.36, "Technical Specifications." This section of 10 CFR requires that each license authorizing operation of a production facility include TSs derived from the analyses and evaluations included in the safety analysis report. This regulation requires the TSs to include items in five specific categories, including (1) safety limits, limiting safety system settings, and limiting control settings, (2) limiting conditions for operation, (3) surveillance requirements, (4) design features, and (5) administrative controls. However, the regulation does not specify any particular requirements to be included in a plant's TSs.

NRC GL 88-16 (Reference 30) was issued to all power reactor licensee's and applicants and discusses removal of cycle-specific parameter limits from TSs to a COLR. This GL allows licensee's to modify their existing TSs in this way provided that the following three conditions are satisfied:

1. The licensee establishes a named formal report (COLR) that includes values of the cycle-specific parameter limits. The cycle specific limits must be determined using an NRC-approved methodology and must be consistent with all applicable limits of the safety analysis.

2. The licensee establishes an administrative reporting requirement to submit the formal report on cycle-specific parameter limits to the Commission for information.

3. The licensee modifies individual TSs to note that cycle-specific parameters shall be maintained within the limits provided in the COLR.

The NRC staff approved the use of a COLR for KNPP through license Amendment No. 165, dated March 11, 2003. The licensee transferred a number of cycle-specific parameters from the TS to the COLR, and demonstrated that all provisions of GL 88-16 were addressed prior to the staff's approval of that amendment request. The current KNPP COLR parameters contain limits associated with Framatome/ANP fuel and the CAOC axial offset control methodology. KNPP is proposing to revise these current COLR parameters to include both the Framatome/ANP parameter value and the corresponding Westinghouse 422V+ parameter value, and to incorporate changes associated with the conversion in axial offset control methodology from CAOC to RAOC. KNPP is not adding any new cycle-specific parameters to the COLR with this amendment request.

To support the COLR changes, the licensee provided a table referencing the NRC-approved methodology used for each of the Westinghouse 422V+ fuel parameter values and the RAOC values being added to the COLR (Reference 33). The NRC staff previously approved each of the topical reports proposed for listing in KNPP TS 6.9 given the conditions described in the reports for appropriate application of the methodologies set forth in them. Each of the methodologies in these reports may be applied at KNPP because the conditions at KNPP under which they would be applied are bounded by the staff-approved conditions set forth in the topical reports, and the licensee has otherwise satisfied the conditions for use set forth in the NRC staff's safety evaluations approving those reports. The NRC staff has reviewed the proposed changes to the KNPP COLR and determined that all provisions of GL 88-16 have been addressed and the proposed COLR changes are acceptable. The NRC staff approved the use of a COLR for KNPP through license Amendment No. 165, dated March 11, 2003.

3.7 <u>TS Changes For Instrumentation and Controls</u>

The following specifications were revised to replace the values of cycle-specific parameter limits with a reference to the COLR that provides these limits.

3.7.1 TS 2.3.a.3.A, Overtemperature ΔT Setpoint

The OTΔT function will trip the reactor when the compensated ΔT in any two channels exceeds the setpoint given by the right hand term in the OTΔT setpoint equation. The KNPP OTΔT setpoint equation is

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

where

ΔT	=	the measured temperature difference between the hot leg and cold leg (°F)
ΔT_0	=	the reference ΔT at 100% rated thermal power (RTP) (°F)
K_1	=	a preset, manually adjustable bias (fraction of RTP ΔT)

K_2 = a preset, manually adjustable constant that accounts for the effects of coolant density and heat capacity on the relationship between ΔT and thermal power (fraction of RTP ΔT/°F)

T = measured average reactor coolant temperature (°F)

T' = reference average reactor coolant temperature at 100% RTP (°F)

τ_1, τ_2 = Laplace lead/lag element time constants (s)

K_3 = a preset, manually adjustable constant to compensate for the effect of RCS pressure on the relationship between ΔT and thermal power (fraction of RTP ΔT/psi)

P = pressurizer pressure (psig)

P' = reference RCS pressure at 100% RTP (psig)

$f(\Delta I)$ = a function of the neutron flux difference between the upper and lower halves of the reactor core (fraction of RTP ΔT)

Although these parameters are incorporated in TS Section 2.3.a.3.A, they are not safety limits, but are used in the OTΔT trip equation to protect against violating safety limits.

The licensee proposed relocating from the TS to the COLR, the manually adjustable values assigned to the applicable OTΔT setpoint equation terms. The licensee stated that these values are expected to change during future refueling cycles as the licensee replaces the existing reactor fuel with Westinghouse 422 VANTAGE + nuclear fuel with PERFORMANCE + features, and as a result of an pending power uprate requested in a January 13, 2003, submittal. This methodology may be applied to 422V+ fuel analyses at KNPP because the conditions at KNPP in which the methodology would be applied are bounded by the conditions set forth in WCAP-14483-A and for which the NRC staff approved the use of the methodology, and the licensee has otherwise satisfied the conditions for use set forth in the NRC staff's safety evaluation approving WCAP-14483-A, dated January 19, 1999. On the basis of its review, the NRC staff concludes that the parameter values are cycle-dependent and are addressed by WCAP-14483-A. The NRC staff concludes, furthermore, that relocation of these parameter values from the TS to the COLR is in accordance with the guidance provided in GL 88-16, as extended by WCAP-14483-A, and is, therefore, acceptable.

3.7.2 TS 2.3.a.3.B, Overpower ΔT Setpoint

The OPΔT function will trip the reactor when the compensated ΔT in any two channels exceed the setpoint given by the right hand term in the OPΔT setpoint equation. The KNPP OPΔT setpoint equation is

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

where

ΔT	=	the measured temperature difference between the hot leg and cold leg (°F)
ΔT_0	=	the reference ΔT at 100% rated thermal power (RTP) (°F)
K_4	=	a preset, manually adjustable bias (fraction of RTP ΔT)
K_5	=	a preset, manually adjustable constant to compensate for piping and thermal delays (fraction of RTP ΔT/°F)
τ_3	=	Laplace rate-lag element time constant (s)
T	=	measured average reactor coolant temperature (°F)
K_6	=	a preset, manually adjustable constant that accounts for the effects of coolant density and heat capacity on the relationship between ΔT and thermal power (fraction of RTP ΔT/°F)
T'	=	reference average reactor coolant temperature at 100% RTP (°F)
$f(\Delta I)$	=	a function of the neutron flux difference between the upper and lower halves of the reactor core, as defined in the TS (fraction of RTP ΔT)

Although these parameters are incorporated in TS Section 2.3.a.3.B, they are not safety limits, but are used in the OPΔT trip equation to protect against violating safety limits.

The licensee proposed relocating from the TS to the COLR the manually adjustable values assigned to the applicable OPΔT setpoint equation terms. The licensee stated that these values are expected to change during future refueling cycles as the licensee replaces the existing reactor fuel with Westinghouse 422 VANTAGE + nuclear fuel with PERFORMANCE + features, and as a result of an pending power uprate requested in a January 13, 2003, submittal. This methodology may be applied to 422V+ fuel analyses at KNPP because the conditions at KNPP in which the methodology would be applied are bounded by the conditions set forth in WCAP-14483-A and for which the NRC staff approved the use of the methodology, and the licensee has otherwise satisfied the conditions for use set forth in the NRC staff's safety evaluation approving WCAP-14483-A, dated January 19, 1999. On the basis of its review, the NRC staff concludes that the parameter values are cycle-dependent and are addressed by WCAP-14483-A. The NRC staff concludes, furthermore, that relocation of these parameter values from the TS to the COLR is in accordance with the guidance provided in GL 88-16, as extended by WCAP-14483-A, and is, therefore, acceptable.

3.8 Revision to TS Section 1.0, "Definition", Item 1P, "Dose Equivalent I-131"

The NRC staff previously accepted this change in Kewaunee License Amendment 166, dated March 17, 2003.

3.9 Changes to TS Section 3.1.c, "Maximum Coolant Activity"

The proposed change will increase DEI-131 limit in the reactor primary coolant to 1.0 µCi/gm from 0.2 µCi/gm for more than 48 hours during one continuous time period, add a new 60 µCi/gm DEI-131 limit for immediate action, delete TS Figure 3.1-3, and revise "10 CFR Part 100 dose guidelines" to "10 CFR 50.67." The NRC staff previously evaluated these requested changes in its radiological consequence analyses for the postulated design-basis accidents in Kewaunee License Amendment 166, dated March 17, 2003, and the NRC staff determined that the changes are acceptable for meeting the dose acceptance criteria specified in 10 CFR 50.67.

3.10 Changes to TS Section 3.4.d, "Secondary Activity Limits"

The proposed change will convert the secondary activity concentration unit from µCi/cc to µCi/gm and to revise "10 CFR Part 100 dose guidelines" to "10 CFR 50.67." The NRC staff finds that these changes are consistent with the NRC staff's radiological consequence evaluation performed in Kewaunee License Amendment 166, dated March 17, 2003, and therefore, the NRC staff finds the proposed changes are acceptable.

3.11 Changes to TS Section 5.2, "Containment"

The proposed change will revise "10 CFR Part 100 dose guidelines" to "10 CFR 50.67." The NRC staff finds that these changes are consistent with the NRC staff's radiological consequence evaluation performed in Kewaunee License Amendment 166, dated March 17, 2003, and therefore, the NRC staff finds the proposed changes are acceptable.

3.12 Changes to TS Table of Contents

The proposed changes are administrative changes for consistency of page numbering. The NRC staff finds these editorial changes acceptable.

3.13 Changes to Bases

The NRC staff finds that the bases changes are consistent with the approved TS changes.

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 56322). 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.

7.0 REFERENCES

1. Letter from M. E. Warner, Nuclear Management Company to USNRC, "License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications, Conforming Technical Specification Changes for Use of Westinghouse VANTAGE + Fuel," Docket No. 50-305, License No. DPR-43, Letter No. NRC-02–067, dated July 26, 2002.

2. Letter from T. Coutu, Kewaunee Nuclear Power Plant to USNRC, "NMC Responses to NRC Request for Additional Information Concerning Licence Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications (TAC No. 5718)," Docket No. 50-305, License No. DPR-43, Letter No. NRC-03–016, dated February 27, 2003.

3. NUREG-0800, "Standard Review Plan," Draft Revision, April 1996.

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

5. Brown, U. L., er al., "PERFORMANCE+ Fuel Features Generic Safety Evaluation," SECL-92-305, dated April 5, 1994.

6. WCAP-12488-P-A, “Westinghouse Fuel Criteria Evaluation Process,” dated October 1994.

7. WCAP-10851-P-A, “Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations,” dated August 1988.

8. WCAP-15063-P-A, “Westinghouse Improved Performance Analysis and Design Model (PAD 4.0),” dated July 2000.

9. WCAP-10125-P-A, “Extended Burnup Evaluation of Westinghouse Fuel,” dated December 1985.

10. WCAP-13589-P-A, “Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel,” dated March 1995.

11. NUREG-1431, Revision 2, “Standard Technical Specifications Westinghouse Plants,” dated June 2001.

12. WCAP-9401-P-A, “Verification Testing and Analysis of the 17x17 Optimized Fuel Assembly,” dated August 1981.

13. WCAP-9273-NP-A, “Westinghouse Reload Safety Evaluation Methodology,” Dated July 1985.

14. WCAP-11596-P-A, “Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores,” dated June 1988.

15. WCAP-10965-P-A, “ANC: A Westinghouse Advanced Nodal Computer Code,” dated September 1986.

16. WCAP-11397-P-A, “Revised Thermal Design Procedure,” dated April 1989.

17. WCAP-8762-P-A, “New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids,” dated July 1984.

18. WCAP-14565-P-A, “VIPRE-01 Modeling and Qualification for Pressurized-Water Reactor Non-LOCA Thermal-Hydraulic Safety Analyses,” dated October 1999.

19. Letter from G. P. Hatchett, USNRC to M. B. Sellman, Wisconsin Electric Power Company, “Point Beach Nuclear Plant, Units 1 and 2 - Issuance of Amendments RE: Design and Operation of Fuel Cycles with Upgraded Westinghouse Fuel (TAC NOS. MA5939 and MA5940),” dated February 8, 2000.

20. 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),” dated December 2002.

21. Letter from M. F. Baumann to USNRC Document Control Desk, “14x14, 0.422" OD VANTAGE + (422V+) Fuel Design,” NPL 97-0538, November 1997.

22. Letter from J. G. Lamb (USNRC) to M. Reddemann (NMC), “Kewaunee Nuclear Power Plant - Review For Kewaunee Reload Safety Evaluation Methods Topical Report WPSRSEM-NP, Revision 3 (TAC NO. MB0306),” dated September 10, 2001.

23. WCAP-7979-P-A, “TWINKLE - A Multidimensional Neutron Kinetics Computer Code,” dated January 1975.

24. WCAP-7908-A, “FACTRAN - A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod,” dated December 1989.

25. WCAP-14882-P-A, “RETRAN-02 Modeling and Qualification for Westinghouse Pressurized-Water Reactor Non-LOCA Safety Analyses,” dated April 1999.

26. WCAP-11394-P-A, “Methodology for the Analysis of the Dropped Rod Event,” dated January 1990.

27. WCAP-7907-P-A, “LOFTRAN Code Description,” dated April 1984.

28. WCAP-7588, Revision 1-A, “An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized-Water Reactors using Spatial Kinetics Methods,” dated January 1975.

29. WCAP-8745-P-A, “Design Basis for the Thermal Overtemperature ΔT and Thermal Overpower ΔT Trip Functions,” dated September 1986.

30. NRC Generic Letter 88-16, “Removal of Cycle-Specific Parameter Limits from Technical Specifications (Generic Letter 88-16),” dated October 3, 1988.

31. WCAP-10216-P-A, Revision 1A, “Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification,” dated February 1994.

32. TSTF 110, Revision 2, approved October 3, 1997.

33. Letter from T. Coutu, KNPP to USNRC, “License Amendment Request 187b to the Kewaunee Nuclear Power Plant Technical Specifications,” Letter No. NRC-03-025, dated March 14, 2003.

34. Letter from T. Coutu, KNPP to USNRC, “License Amendment Request No. 187 to the Kewaunee Nuclear Power Plant Technical Specifications,” Letter No. NRC-03-032, dated March 21, 2003 (KNPP letter discussing analysis and results using WCAP-12488).

35. Letter from T. Coutu, KNPP to USNRC, “Response to Request For Additional Information Regarding License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications,” Letter No. NRC-03-035, dated March 21, 2003 (KNPP letter on WCAP-7588 Rod Ejection results applicability).

36. S.I. Deterer, et al., WCAP-14449-P-A, “Application of Best-Estimate Large-Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection, Rev. 1,” October 1999.

37. WCAP-13677-P-A, 10CFR50.46 Evaluation Model Report: WCOBRA/TRAC Two-Loop Upper Plenum Injection Model Updates to Support ZIRLO^{TM} Cladding Option,” April 1993, as approved in an NRC Safety Evaluation Report dated November 26, 1993.

38. C.M. Thompson, et al., “Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model,” WCAP-10054-P-A, Addendum 2, Revision 1, dated July, 1997.

39. 10 CFR 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors.” (As of January 1, 2002).

40. 10 CFR Part 50, Appendix A, General Design Criterion 35, “Emergency Core Cooling” (As of January 1, 2002).

41. 10 CFR Part 50, Appendix K, “ECCS Evaluation Models” (As of January 1, 2002).

42. Letter from T. Coutu, KNPP to USNRC, “License Amendment Request No. 187 to the Kewaunee Nuclear Power Plant Technical Specifications,” Letter No. NRC-03-031, dated March 19, 2003.

Principal Contributors: M. Kowal, C. Liang
A. Attard F. Orr
M. Waterman J. Lee

Date: April 4, 2003