UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 1, 2009

Mr. Michael D. Wadley
Site Vice President
Prairie Island Nuclear Generating Plant
Northern States Power - Minnesota
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: TECHNICAL SPECIFICATIONS CHANGES TO ALLOW USE OF WESTINGHOUSE 0.422-INCH OD 14X14 VANTAGE+ FUEL (TAC NOS. MD9142 AND MD9143)

Dear Mr. Wadley:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 192 to Facility Operating License No. DPR-42 and Amendment No. 181 to Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Units 1 and 2 (PINGP), respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated June 26, 2008, as supplemented by letters dated August 4, August 26, and November 14, 2008, and January 30, February 9, February 20, March 12, and May 4 (2 letters), 2009.

The amendments revise the PINGP TSs for the use of Westinghouse 422 VANTAGE+ nuclear fuel and make changes to certain references identified in the Design Features section of the TSs.

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

Sincerely,

Thomas J. Wengert

Thomas J. Wengert, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosures:
1. Amendment No. 192 to DPR-42
2. Amendment No. 181 to DPR-60
3. Safety Evaluation

cc w/encls: Distribution via ListServ

NORTHERN STATES POWER COMPANY*

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 192
License No. DPR-42

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

   A. The application for amendment by Nuclear Management Company, LLC* (the licensee), dated June 26, 2008, as supplemented by letters dated August 4, August 26, and November 14, 2008, and January 30, February 9, February 20, March 12, and May 4 (2 letters), 2009, 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-42 is hereby amended to read as follows:

* On September 22, 2008, Nuclear Management Company, LLC (NMC), transferred its operating authority to Northern States Power Company, a Minnesota Corporation (NSPM). By letter dated September 3, 2008, NSPM stated that it would assume responsibility for actions and commitments submitted by NMC.

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 192 , are hereby incorporated in the license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Peter S. Tam for

Lois M. James, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility Operating License and Technical Specifications

Date of Issuance: July 1, 2009

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY*

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 181
License No. DPR-60

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

   A. The application for amendment by Nuclear Management Company, LLC* (the licensee), dated June 26, 2008, as supplemented by letters dated August 4, August 26, and November 14, 2008, and January 30, February 9, February 20, March 12, and May 4 (2 letters), 2009, 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-60 is hereby amended to read as follows:

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* On September 22, 2008, Nuclear Management Company, LLC (NMC), transferred its operating authority to Northern States Power Company, a Minnesota Corporation (NSPM). By letter dated September 3, 2008, NSPM stated that it would assume responsibility for actions and commitments submitted by NMC.

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 181 , are hereby incorporated in the license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Peter S. Tam for

Lois M. James, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility Operating License and Technical Specifications

Date of Issuance: July 1, 2009

ATTACHMENT TO LICENSE AMENDMENT NOS. 192 AND 181

FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60

DOCKET NOS. 50-282 AND 50-306

Replace the following pages of the Facility Operating License No. DPR-42 and DPR-60 with the attached revised pages. The changed areas are identified by a marginal line.

REMOVE	INSERT
DPR-42, License Page 3	DPR-42, License Page 3
DPR-60, License Page 3	DPR-60, License Page 3

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

REMOVE	INSERT
2.0-1	2.0-1
3.5.1-2	3.5.1-2
4.0-2	4.0-2
4.0-3	4.0-3
4.0-4	4.0-4

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, NSPM to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility;

(6) Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to transfer byproduct materials from other job sites owned by NSPM for the purpose of volume reduction and decontamination.

C. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

NSPM is authorized to operate the facility at steady state reactor core power levels not in excess of 1650 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 192 , are hereby incorporated in the license. NSPM shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

NSPM shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Prairie Island Nuclear Generating Plant Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program," Revision 1, submitted by letters dated October 18, 2006, and January 10, 2007.

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility;

(6) Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to transfer byproduct materials from other job sites owned by NSPM for the purposes of volume reduction and decontamination.

C. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

NSPM is authorized to operate the facility at steady state reactor core power levels not in excess of 1650 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 181 , are hereby incorporated in the license. NSPM shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

NSPM shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Prairie Island Nuclear Generating Plant Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program," Revision 1, submitted by letters dated October 18, 2006, and January 10, 2007.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR and the following SLs shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained $\geq$ 1.17 for WRB-1 DNB correlation for OFA fuel.

2.1.1.2 The peak fuel centerline temperature shall be maintained as follows:
a) < 5080 °F, decreasing by 58 °F per 10,000 MWD/MTU burnup, for fuel containing UO_2;
b) < (5080 °F minus 6.75 °F per w/o Gd_2O_3), decreasing by 58 °F per 10,000 MWD/MTU burnup, for fuel containing gadolinia.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained $\leq$ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two accumulators inoperable.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	12 hours
SR 3.5.1.2	Verify borated water volume in each accumulator is ≥ 1250 cubic feet (25%) and ≤ 1290 cubic feet (91%).	12 hours
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ≥ 710 psig and ≤ 770 psig.	12 hours
SR 3.5.1.4	Verify boron concentration in each accumulator is ≥ 2300 ppm.	31 days
SR 3.5.1.5	Verify power is removed from each accumulator isolation valve operator when RCS pressure is ≥ 2000 psig.	31 days

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in USAR Section 10.2;

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 730 ppm, which includes an allowance for uncertainties as described in USAR Section 10.2;

d. A nominal 9.5 inch center to center distance between fuel assemblies placed in the fuel storage racks;

e. New or spent fuel assemblies with a combination of discharge burnup, initial enrichment and decay time in the "unrestricted range" of Figure 3.7.17-1 may be allowed unrestricted storage in the fuel storage racks; and

f. New or spent fuel assemblies with a combination of discharge burnup, initial enrichment and decay time in the "restricted range" of Figure 3.7.17-1 will be stored in compliance with Figures 4.3.1-1 through 4.3.1-4.

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. keff < 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties as described in USAR Section 10.2;

c. $k_{eff} \leq 0.98$ if accidentally filled with a low density moderator which resulted in optimum low density moderation conditions; and

d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.1.3 Fuel will not be inserted into a TN-40 spent fuel cask in the pool unless a minimum boron concentration of 1800 ppm is present. The 1800 ppm will ensure that k_{eff} for the spent fuel cask, including statistical uncertainties, will be ≤ 0.95 for all postulated arrangements of fuel within the cask. The criticality analyses for the TN-40 spent fuel storage cask were based on fresh fuel enriched to 3.85 weight percent U-235.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 727′ 4″ (Mean Sea Level).

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1386 fuel assemblies not including those assemblies which can be returned to the reactor. The southeast corner of the small pool serves as the spent fuel cask lay down area. To facilitate plant evolutions, four additional storage racks, with a combined capacity of 196, may be temporarily installed in the cask lay down area to provide a total of 1582 storage locations (USAR Section 10.2).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 192 TO FACILITY OPERATING LICENSE NO. DPR-42

AND AMENDMENT NO. 181 TO FACILITY OPERATION LICENSE NO. DPR-60

NORTHERN STATES POWER COMPANY - MINNESOTA

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By application dated June 26, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML081820137), as supplemented by letters dated August 4 (ADAMS Accession No. ML082210260), August 26 (ADAMS Accession No. ML082400518), and November 14, 2008 (ADAMS Accession No. ML083190818), and January 30 (ADAMS Accession No. ML090300684), February 9 (ADAMS Accession No. ML090700271), February 20 (ADAMS Accession No. ML090510691), March 12 (ADAMS Accession No. ML090721087), and May 4 (2 letters; ADAMS Accession Nos. ML091240262 and ML091310384), 2009, Nuclear Management Company, LLC, a predecessor license holder to Northern States Power Company, a Minnesota corporation (NSPM, the licensee), requested changes to the Technical Specifications (TSs) for the Prairie Island Nuclear Generating Plant, Units 1 and 2 (PINGP). The proposed changes would revise the PINGP TSs to accommodate the use of Westinghouse 422 VANTAGE+ nuclear fuel, and make certain changes to references identified in the Design Features section of the TSs.

The licensee has analyzed or evaluated the nuclear and fuel design, thermal-hydraulic design, re-evaluated the accident and transient analyses, some of which were re-analyzed, and the loss-of-coolant accident (LOCA) and non-LOCA analyses that were affected by the requested fuel transition.

The supplemental information dated August 4, August 26, and November 14, 2008, and January 30, February 9, February 20, March 12, and May 4 (2 letters), 2009, contained clarifying information, did not change the scope of the June 26, 2008, 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 Nuclear Regulatory Commission (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 mechanical analysis. Each of these subjects is evaluated separately in the respective sections which follow.

Pressurized-Water Reactor (PWR) Matrix 8 of Regulatory Standard 001 (RS-001), "Review Standard for Extended Power Uprates," provides a standard regulatory evaluation and review procedure for, among other things, the fuel system thermal hydraulic design and the accident and transient analyses (Reference 8). Therefore, the NRC staff used RS-001, PWR Matrix 8, to conduct its detailed review of this license amendment request.

The U.S. Atomic Energy Commission (AEC) issued a "Safety Evaluation of the Prairie Island Nuclear Generating Plant" on September 28, 1972, supplemented March 21, 1973, April 30, 1973, and May 31, 1973. PINGP is not a General Design Criteria (GDC) plant. However, the AEC performed a technical review of PINGP against the GDC in effect at the time and concluded that the PINGP design generally conforms to the intent of 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

The fuel system consists of arrays of fuel rods, burnable poison rods, spacer grids and springs, end plates, and reactivity control rods. The NRC staff reviewed the fuel system to ensure that:

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

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

(3) The number of fuel rod failures is not underestimated for postulated accidents, and

(4) Coolability is always maintained.

The NRC staff's review covered 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) Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50.46, insofar as it establishes the standard for the calculation of emergency core cooling system (ECCS) performance and acceptance criteria for that calculated performance;

(2) GDC 10, insofar as it requires that the reactor core be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of AOOs;

(3) GDC 27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with

appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and

(4) GDC 35, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA.

Specific review criteria are contained in Section 4.2 of the Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (SRP), and additional guidance contained in Matrix 8 of RS-001.

2.1.B Fuel System Design Technical Evaluation

The PINGP cores contain Westinghouse VANTAGE+ fuel design with a 14x14 array. The licensee stated that it will plan a fuel transition from the current 0.400-inch outer diameter (OD) VANTAGE+ (hereafter referred to as 400V+) fuel to 0.422-inch OD VANTAGE+ (hereafter referred to as 422V+) fuel commencing with Cycle 26 for both Units 1 and 2. The 422V+ fuel is an upgraded Westinghouse fuel design with PERFORMANCE+ features. The VANTAGE+ fuel design features the ZIRLO™ cladding and was approved in a topical report WCAP-12610-P-A, entitled "VANTAGE+ Fuel Assembly Reference Core Report" (Reference 9). The PERFORMANCE+ features can include pre-oxidized cladding, a protective bottom grid, ZIRLO™ mid-grids and intermediate flow mixing grids, and low cobalt top and bottom nozzles. The licensee stated that the primary fuel change is a change in diameter from 0.400 to 0.422 inches. The licensee asserted that the 422V+ fuel assembly is designed to be mechanically and hydraulically compatible with the current existing Westinghouse fuel in full or transition cores. The 422V+ fuel design is currently in use at Kewaunee Nuclear Power Plant (KNPP) and Point Beach Nuclear Plant Units 1 and 2. The licensee performed the following analyses for the 422V+ fuel design using the currently existing fuel rod design criteria with a fuel performance code PAD 4.0. The PAD 4.0 code is an approved Westinghouse fuel performance code, as described in a topical report WCAP-15063-P-A, entitled "Westinghouse Improved Performance Analysis and Design Model" (PAD 4.0) (Reference 10).

2.1.B.1 NRC Staff Position

In the past, the NRC staff has approved various burnup limits for Westinghouse fuel designs and fuel performance codes. Noticeably, the VANTAGE fuel series including VANTAGE+ was approved to the burnup limit of 60,000 megawatt days per metric ton uranium (MWD/MTU), and the more recent Westinghouse fuel performance code, PAD 4.0, was approved to the burnup limit of 62,000 MWD/MTU.

Recently, the NRC staff conducted an audit of the Westinghouse documents describing the fuel data, analytical models, and the fuel change procedures. The NRC staff reviewed documents including several plant reload analyses. The reload analyses provided results for all the SAFDLs as described in 10 CFR 50, Appendix A, GDC 10. The analyses were typically performed at bounding conditions such that plant thermal-mechanical safety evaluations (SEs) were not required for each reload cycle. The SAFDLs include rod internal pressure, clad stress and strain, corrosion, clad fatigue, fuel melting temperature, rod growth, creep collapse, etc. The results showed that all SAFDLs were met for the bounding conditions. The NRC staff also recognized that the SAFDLs were analyzed using the PAD 4.0 code which was approved to a peak rod average burnup of 62,000 MWD/MTU. Based on the audit results, the NRC staff concludes that burnup limit for WCAP-12610-P-A can be increased to 62,000 MWD/MTU provided that the evaluation of the fuel design performance is performed with PAD 4.0.

By letter dated May 25, 2006 from J. D. Peralta (NRC) to B. F. Maurer (Westinghouse), the NRC staff approved the burnup increase for the VANTAGE fuel series (Reference 11). Therefore, the burnup limit of the VANTAGE+ fuel design, including the 422V+ fuel, can be increased to 62,000 MWD/MTU peak rod average.

2.1.B.2 Rod Internal Pressure

The licensee stated that the rod internal pressure will be limited to a value below that which could cause the closed fuel-to-clad diametral gap to re-open and move outward due to the cladding creep, thereby inducing extensive departure from nucleate boiling (DNB) propagation. Extensive DNB propagation could result in severe fuel failures and core damage.

The licensee analyzed the rod pressure using the PAD 4.0 code and determined that the rod pressure meets the design limit of no gap re-opening. Furthermore, the licensee's analysis showed that the 422V+ fuel does not experience extensive DNB propagation. That is, the DNB propagation process, if it occurs, is self-limiting and results in very small fuel failures and no core damage under transient conditions. Based on the licensee's analysis using the existing methodology, the NRC staff concludes that the rod pressure analysis is acceptable for the 422V+ fuel design.

2.1.B.3 Cladding Oxidation and Hydriding

The licensee stated that the cladding metal-to-oxide interface temperature is required to be below specified limits in order to prevent accelerated oxidation, and cladding hydrogen pickup is required to be less than or equal to 600 parts per million (ppm) on a volumetric average basis at end of life (EOL). Excessive oxidation or hydrogen pickup could lead to cladding premature failure.

The licensee analyzed the cladding metal-to-oxide interface temperature using the PAD 4.0 code and determined that the interface temperatures under steady-state and transient conditions are all below the specified limits. The licensee also analyzed the hydrogen pickup rate and determined that the total hydrogen pickup meets the 600 ppm requirements at EOL. Based on the licensee's analyses using the existing methodology, the NRC staff concludes that the cladding oxidation and hydriding analyses are acceptable for the 422V+ fuel design.

2.1.B.4 Fuel Temperature

The licensee stated that the calculated fuel centerline temperature is required not to exceed the fuel melting temperature. Fuel melting could result in fuel failures.

The licensee analyzed the fuel pellet temperature using the PAD 4.0 code and determined that fuel centerline temperature is below the melting temperature under steady-state and transient conditions. Based on the licensee analysis using the existing methodology, the NRC staff concludes that the fuel temperature analysis is acceptable for the 422V+ fuel design.

2.1.B.5 Cladding Flattening

The licensee stated that the fuel rod is designed to prevent cladding flattening due to long-term creep collapse. The cladding flattening could occur if significant axial gaps exist in the fuel column due to fuel densification. A flattening cladding is considered failed.

The licensee stated that the approved WCAP-13589-A, entitled "Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel," provides justification to confirm that axial gaps in the fuel column will not occur for Westinghouse fuel designs (Reference 12). The licensee confirmed that the 422V+ fuel design meets the criteria in WCAP-13589-A. Based on the licensee's analysis using the existing methodology, the NRC staff concludes that the cladding flattening analysis is acceptable for the 422V+ fuel design.

2.1.B.6 Fuel Rod Axial Growth

The licensee stated that the fuel rod is designed to provide sufficient axial space to accommodate expected fuel rod growth without degradation of the assembly function. Excessive rod or assembly bowing could occur if the clearance between the fuel rod and the top and bottom nozzles cannot be maintained due to fuel rod growth. Rod or assembly bowing could result in impact in the thermal-hydraulic safety analysis.

The licensee analyzed fuel rod growth using the PAD 4.0 code and determined that there is adequate clearance for fuel rod growth to prevent contact between the fuel rod and the top and bottom nozzles. Based on the licensee's analysis using the existing methodology, the NRC staff concludes that the fuel rod growth analysis is acceptable for the 422V+ fuel design.

2.1.B.7 Seismic/LOCA Impact

The licensee stated that the fuel assembly is designed to maintain its structural integrity under combined seismic and LOCA loading conditions. Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assemblies. The damage from these external forces could prevent control rod insertion when the reactor shutdown is required. In Enclosure 1, Attachment 4, of its application, the licensee stated that evaluations have demonstrated that the core coolable geometry and control rod insertion requirements are met.

In a letter responding to the NRC staff's request for additional information (RAI) dated August 4, 2008 (Reference 2), the licensee indicated that the seismic and LOCA loading analysis was performed using the approved methodology described in WCAP-9401-P-A, entitled "Verification Testing and Analysis of the 17x17 Optimized Fuel Assembly" (Reference 13). The licensee analyzed the homogeneous core of 422V+ fuel and the mixed core of 422V+ and 400V+ fuel. The maximum grid impact forces from a safe shutdown earthquake and LOCA are combined using the square root of the sum of the square (SRSS) method. The licensee determined that the maximum forces occur in the peripheral assemblies for mixed core conditions. However, the results showed that the forces are less than the allowable grid spacer strength, resulting in no deformed grid spacers and thus maintaining the structural integrity. Thus, the core coolable geometry and control rod insertion requirements are met.

Based on the licensee's analyses using the approved methodology, the NRC staff concludes that the seismic and LOCA loading analysis is acceptable for the 422V+ fuel design.

2.1.C Fuel System Design Conclusion

The NRC staff reviewed the licensee's mechanical design analyses. Based on the licensee's analyses using the existing methodologies including the PAD 4.0 code, the NRC staff concludes that the 422V+ fuel mechanical design is acceptable to the peak rod average burnup limit of 62,000 MWD/MTU for PINGP.

2.2 Nuclear Design Evaluation

2.2.A Nuclear Design Regulatory Evaluation

The NRC staff reviewed 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 and 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 covered core power distribution, reactivity coefficients, reactivity control requirements and control provisions, control rod patterns and reactivity worths, criticality, burnup, and vessel irradiation.

The NRC's acceptance criteria are based on:

(1) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs;

(2) GDC-11, insofar as it requires that the reactor core be designed so that the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity;

(3) GDC-12, insofar as it requires that the reactor core be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can be reliably and readily detected and suppressed;

(4) GDC-13, insofar as it requires that instrumentation and controls be provided to monitor variables and systems affecting the fission process over anticipated ranges for normal operation, AOOs and accident conditions, and to maintain the variables and systems within prescribed operating ranges;

(5) GDC-20, insofar as it requires that the protection system be designed to initiate the reactivity control systems automatically to assure that acceptable fuel design limits are not exceeded as a result of AOOs and to automatically initiate operation of systems and components important to safety under accident conditions;

(6) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems;

(7) GDC-26, insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes;

(8) GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and

(9) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core.

Specific review criteria are contained in SRP Section 4.3 and other guidance provided in Matrix 8 of RS-001.

2.2.B Nuclear Design Technical Evaluation

The licensee stated that there are no changes to the Westinghouse nuclear design methodology for the transition to the 422V+ fuel design due to the similar design features of the 400V+ and 422V+ fuel assemblies. The licensee used standard nuclear design methodology, previously used for 400V+ fuel, to analyze the 422V+ fuel design. The results showed that the radial peaking factor and total peaking factor, 1.77 and 2.50 respectively, remain the same for the 422V+ fuel design. The analysis also showed that key safety parameters such as DNB boiling, peak cladding temperature, and peak linear heat rate remain the same for the 422V+ fuel design. Based on the licensee's analyses using the standard nuclear design methodology, the NRC staff concludes that the 422V+ fuel nuclear design is acceptable for PINGP.

2.2.C Nuclear Design Conclusion

The NRC staff has reviewed the licensee's analyses related to the effect of the proposed fuel upgrade on the nuclear design of the fuel assemblies, control systems, and reactor core. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed fuel upgrade 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 upgrade 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 reviewed 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 is a justified extrapolation from proven designs,

(3) Provides acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation and anticipated operational occurrences, and

(4) Is not susceptible to thermal-hydraulic instability.

The NRC review also covers hydraulic loads on the core and RCS components during normal operation and design-basis accident conditions and core thermal-hydraulic stability under conditions of normal operation, AOOs, and anticipated transients without scram.

The NRC staff's acceptance criteria are based on

(1) GDC 10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; and

(2) GDC 12, insofar as it requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can be reliably and readily detected and suppressed.

Specific review criteria are contained in Section 4.4 of the SRP and other guidance provided in Matrix 8 of RS-001.

2.3.B Thermal and Hydraulic Design Technical Evaluation

The licensee stated that the thermal hydraulic analysis of the 14x14 422V+ fuel at PINGP is based on the Revised Thermal Design Procedure (RTDP), which is documented in Reference 14, and the use of the WRB-1 DNB correlation. To support the requested change, the licensee evaluated the limits applicable to both the DNB ratio (DNBR) and the DNB correlation, as well as the RTDP. The licensee also accounted for transition core effects and thermal mechanical effects on the thermal and hydraulic design.

2.3.B.1 Evaluation of Revised Thermal Design Procedure

The RTDP methodology statistically accounts for the system uncertainties in plant operating parameters, fabrication parameters, nuclear and thermal parameters, as well as the DNB correlation and computer code uncertainties. The RTDP is used to establish a design DNBR limit that statistically accounts for the effects of the key parameters on DNB.

According to the NRC staff's approval SE for Reference 14, the RTDP was developed for predicting the DNBR design limit for PWRs. The NRC staff reviewed the conditions and limitations set forth in the SE, and determined that the requested fuel transition does not affect compliance with any of the conditions and limitations. Therefore, the use of the RTDP is acceptable with respect to the requested fuel upgrade, because it is an NRC-approved method that is appropriate for use with the upgraded fuel.

2.3.B.2 Evaluation of Change in Analytic Flow Uncertainty

The licensee stated in Chapter 1 of the Licensing Report (LR) [Enclosure 1, Attachment 4 of Reference 1]:

> The power uncertainty was reduced to account for installation of a more accurate flow measurement system used in the power measurement. The RTDP analyses completed within this report were thus completed at a bounding high power level to confirm

acceptable operation at any power level, including measurement uncertainties of 0.5 percent or more, up to 1,683 MWt.

While this change in uncertainty will cause results to be effectively the same as that obtained using the previous analytic method, because the power uncertainty is accounted for by use of a bounding power level rather than a parametric uncertainty parameter, the NRC staff believed that the change should be justified based on the change in operations resulting in this uncertainty reduction. Therefore, the NRC staff requested that the licensee:

1. Explain what flow measurement system was installed;

2. Provide reference to applicable supporting documentation, such as topical reports describing the flow measurement system; and

3. Briefly describe the flow measurement system installation and calibration process.

In the RAI response letter (Reference 3), the licensee stated that the uncertainty re-allocation was performed based on the installation of the Leading Edge Flow Meter (LEFM) CheckPlus system. The system is supported by NRC-approved topical reports ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM System" (Reference 15), and ER-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System" (Reference 16). The licensee also described the LEFM system installation and stated that it was calibrated at Alden Laboratories in unit-specific piping configurations.

The NRC staff requested this additional information to establish the necessary justification for re-allocating uncertainties from the RTDP. In its introductory passage, the licensee clarified that all necessary licensing basis uncertainties in the core power level are retained, but that an element of the statistical uncertainty is reduced, and then recovered in the deterministic uncertainty. The NRC staff agrees that, altogether, the necessary 2 percent calorimetric power uncertainty is retained in the RTDP transient analyses.

The NRC staff sought justification of the uncertainty re-allocation to ensure that the RTDP uncertainties in use were based on judicious and justified, rather than capricious and arbitrary, selection of uncertainty parameters. In response to Questions 1 and 2 above, the licensee explained that the RTDP uncertainty reduction is based on the installation of the Cameron Leading Edge Flow Meter ultrasonic flow monitoring device.

In the NRC staff's experience, this device can have the capability to perform flow measurements to within 0.36-percent accuracy, which makes the 0.5-percent RTDP uncertainty assumed conservative and bounding of the device's understood measuring capability. The licensee also confirmed that the 0.36-percent capability has been calculated for both units. The licensee referenced NRC staff-approved licensing topical reports that support the device's installation and measuring capability. The licensee also stated, in response to Question 3, that the units' stated flow measuring accuracy is supported by laboratory calibration, and confirmed to be applicable based on in-situ testing.

Based on the following considerations: (1) the proposed RTDP power uncertainty is bounding of the known capability of the flow measurement device to be installed at PINGP; (2) the required additional flow measurement uncertainty will be incorporated into the transient analyses deterministically; and (3) the proposed flow measurement device is supported by NRC-approved licensing topical reports, laboratory calibration, and in-situ confirmation of the calibration, the NRC staff concludes that the RTDP uncertainty re-allocation proposed by the licensee is justified.

The NRC staff also observed that the discussion in Chapter 1 of the LR regarding the RTDP uncertainty reduction appeared to be inconsistent with a statement in Chapter 4 of the LR, "A power level of 1,677 MWt has been used for all RTDP thermal-hydraulic design analyses. For analyses explicitly modeling parameter uncertainties, a power level of 1,683 MWt was used." The NRC staff requested clarification regarding the apparent differences between the information contained in Chapters 1 and 4 of the LR.

The licensee clarified that the 1,677 MWt power level, with a 0.5 percent statistical power uncertainty, bounds operation at 1650 MWt with a total 2-percent uncertainty in the core power level to account for calorimetric uncertainty. Because the NRC staff understands that the RTDP is a procedure that requires the statistical combination of uncertainties, there was a concern that the total power uncertainty had not been accounted for in a manner that was the same as, or conservative relative to, the typical method set forth in Reference 14.

The NRC staff requested that the licensee perform a comparative study to demonstrate the conservatism of the analytic uncertainty treatment at the current licensed thermal power level when compared to using the RTDP. The licensee performed these studies and described the results in Reference 4. The conclusion was that the DNBR limits, when analyzed using a fractional deterministic power uncertainty with the remainder allocated statistically, resulted in DNBR limits that were conservative relative to analyses that allocate all power uncertainty statistically. The NRC staff reviewed this response and accepted it because it demonstrates that the licensee's proposed approach is conservative relative to that previously accepted by the NRC staff and described in Reference 14.

Based on the NRC staff's evaluation of the licensee's re-allocation of the power level uncertainty as analyzed using the RTDP, the NRC staff concludes that the licensee's approach to split the uncertainty into a deterministic component and treat the rest statistically is acceptable. While the approach is reflective of the implementation of a planned improvement, the licensee has demonstrated that the approach is conservative when compared to an analysis that is reflective of the current plant configuration.

2.3.B.3 Evaluation of DNBR Limits and Margin

The DNB design criterion reflects the guidance contained in Chapter 4.4 of the SRP, specifically, that the appropriate margin is contained in the RTDP statistical analysis to provide 95/95 confidence that the limiting fuel rods will not undergo transition boiling as discussed in the preceding paragraphs. As the RTDP considers the parametric uncertainties, thermal-hydraulic analyses are performed using input parameters at their nominal values.

A safety analysis limit (SAL) DNBR is calculated, which provides for a certain amount of margin above the design limit discussed above. The SAL, which is higher than the design limit, provides a margin to offset the effect of rod bow and other DNBR penalties that may occur, and also provides the licensee with margin for operational flexibility.

The licensee is implementing the RTDP using the WRB-1 DNB correlation. The licensee stated that use of the WRB-1 correlation for cores consisting of a mixed loading of 422V+ and 400V+ has been shown to be valid using the WRB-1 correlation, as documented in an NRC staff SE report approving the use of the WRB-1 DNB correlation for use in 14x14 fuel (Reference 17) and in the NRC-approved topical report describing the VIPRE-01 modeling and qualification (Reference 18).

The NRC staff confirmed that Reference 17 provides for application of WRB-1 to 14x14 fuel assemblies with 0.422-inch OD fuel assemblies, based on bundle tests using the same fuel design. The staff found that the WRB-1 correlation is acceptable for application to 14x14 Optimized Fuel Assemblies (OFA) with a minimum DNBR limit of 1.17. In its SE for Reference 18, the NRC staff found that VIPRE, Westinghouse's current generation transient thermal hydraulic modeling tool, is acceptable for use for an explicitly delineated set of DNB correlations. WRB-1 is included in this set, and was again approved for modeling with a correlation limit of 1.17.

The NRC staff's approval of the WRB-1 applicability to 14x14 fuel designs is based on testing results of the OFA, and the licensee proposes to install VANTAGE+ (V+) fuel assemblies. While having similar lattice geometry, the grid structures are different, and the differences have the potential to affect DNB performance. Westinghouse evaluated the potential effects and affirmed its conclusion that the DNB correlation limit of 1.17 can be supported for the V+ fuel design. The evaluations supporting WRB-1 applicability to 14x14, 0.422-inch OD fuel VANTAGE+ fuel are referenced in a recent Fuel Criterion Evaluation Process Notification concerning design changes to the 14x14 V+ fuel design (References 19 and 20). Because Westinghouse's evaluations were performed in accordance with an NRC-approved design and notification process, the NRC staff concludes that the 1.17 limit, originally approved for the OFA fuel design, is acceptable for the V+ fuel design, as well.

Not all transients are analyzed using the WRB-1 correlation and/or the RTDP. For those transients where use of the WRB-1 correlation or the RTDP is restricted, the licensee used the Standard Thermal Design Procedure (STDP) and/or the W-3 DNB correlation, as appropriate. These transients are those for which the analyzed conditions are predicted to fall outside the applicability range of the WRB-1 correlation or the RTDP based on of pressure, local mass velocity, bypass conditions, or power peaking. Any departures from the use of WRB-1 and/or the RTDP are noted in the appropriate sections of the LR, and in the NRC staff's SE.

After setting the safety analysis limit DNBR, the licensee uses the SAL to develop core limits, axial offset limits, and dropped rod limits. The maximum enthalpy rise hot channel factor is then developed based on these limits. As discussed in the LR, this limit is 1.77 for the 422V+ fuel.

The licensee stated that the DNBR margin evaluation, performed as described above, and summarized in Table 4.4 of the LR, is cycle-dependent and may vary from that documented in the LR for future reload designs. The DNBR margins are evaluated in accordance with NRC-approved methodology (Reference 30), and on this basis, the analytical technique is acceptable to the NRC staff.

2.3.B.4 Evaluation of Hydraulic Compatibility and Transition Core Effects

As a result of transition core designs employing mixed loadings of 422V+ and 400V+ fuel, there will be an amount of flow redistribution among the fuel assemblies. The licensee stated that differences in hydraulic resistance among the fuel assemblies will become an additional mechanism for flow redistribution, resulting in a crossflow component to the normally dominant axial flow direction. The licensee stated that this could affect the fuel assembly mechanical design in two ways: (1) the peripheral rods in the fuel assemblies could be prone to fretting or whirling wear mechanisms, and (2) the assembly lift forces could be influenced by the introduction of higher resistance assemblies. These issues are addressed in Section 2.1 of this SE.

Crossflow will, however, affect both the LOCA and the DNB analyses. For the LOCA analyses, the core will experience a reduction in the normalized mass velocity as compared to a full core of a given assembly type. For DNB analyses, the flow redistribution will affect the predictions of minimum DNBR. To account for this, a transition DNBR penalty is applied to account for the phenomenology. The licensee stated that transition cores are analyzed as if they were full cores of one assembly type by applying the applicable transition core penalty. The calculation of a penalty is described in an NRC-approved methodology document (Reference 22). The transition core DNBR penalty for the 400V+ fuel is, according to the licensee, offset by a proportional sensitivity to radial power peaking. The licensee stated, "There is a maximum 9.0-percent transition core DNBR penalty for the 400V+ fuel which will be offset by a 6.0-percent FdH reduction in burned 400V+ fuel based on a conservative 1.5-percent DNBR: 1-percent FdH sensitivity."

This treatment of DNBR margin trade-off was presented as axiomatic; however, the NRC staff was unfamiliar with this sensitivity, and hence requested additional information regarding the basis for this statement. Specifically, the NRC staff requested that the licensee:

1. Reference an appropriate licensing topical report where this sensitivity is described

2. Provide a phenomenological discussion of the peaking behavior of previously irradiated fuel and explain how the changes in peaking behavior result in increased DNBR margin

3. Clarify whether the DNBR margin tradeoff occurs during steady-state or transient situations

The licensee's response is contained in Reference 3. For Question 1, the licensee referred the NRC staff to WCAP-11397, Reference 14, which the NRC staff reviewed to find the applicable discussion. Although the NRC staff did not locate the relevant discussion, the licensee provided sufficient phenomenological discussion in response to Question 2 to resolve the NRC staff's concern.

In response to Question 2, the licensee stated:

> The peaking factor restriction is on the burned 400V+ fuel. At increasing burnups (second and third resident cycle in the core), the build-up of fission products and the depletion of fissile material results in the burned fuel assemblies no longer being able to maintain a high peaking factor. The peaking factor reduction in the burned 400V+ fuel necessary to offset the transition core DNBR penalty on the 400V+ fuel is conservatively calculated using the stated sensitivity factor and is then confirmed during the reload core design for each transition cycle.

For gadolinia-bearing fuel and cores designed with significant loadings of burnable absorber assemblies, the burnout of the fuel doping and burnable absorber can tend to compensate for the fissile depletion and fission product build-up identified above. To confirm that these phenomena do not self-compensate to a degree that would cause a safety concern, the NRC staff evaluated the first transition cycle peaking factors contained in Figure 3-2 of the LR to compare differences in beginning-of-cycle peaking factors between the feed fuel and the once-burnt resident fuel. Indeed, the highest peaking factor of the once-burnt fuel was more than nine percent lower than the highest peaking factor of the fresh fuel. Using this evaluation, the NRC staff confirmed that the peaking factor reduction assumed for the first cycle reference transition core is conservative. Because this assumption will be validated as a part of the NRC-approved, cycle-specific reload process as noted in the licensee's response, this DNBR penalty trade-off is acceptable based on the peaking factor reduction assumption.

In response to Question 3, the licensee stated that initial conditions are chosen such that the axial and radial peaking factors bound transient power distribution changes anticipated during the transient. Therefore, the NRC staff finds the peaking factor trade-off is acceptable for transient analyses because the transient analyses use steady-state initial conditions intended to bound the worst case transient power distribution changes.

2.3.B.5 Evaluation of Thermal-Mechanical Effects on Fuel Thermal Hydraulics

A burnup-dependent penalty has been applied to the DNB limits to account for rod bowing between grid spacers. These penalties are 2.9-percent DNBR for the 422V+ fuel and 3.0-percent for the 400V+ fuel. The licensee stated that, for burnups greater than 24,000 MWD/MTU, credit is taken for a power peaking burndown due to the decrease in fissionable isotopes and the buildup of fission product inventory. The licensee stated, therefore, that no additional rod bow penalty is required at burnups greater than 24,000 MWD/MTU.

The NRC-approved Westinghouse code PAD 4.0 predicts fuel temperatures and associated rod internal pressures for the gadolinium-bearing 422V+ and 400V+ fuel types, as documented in Reference 10. The licensee stated that, owing to a lower thermal conductivity, the gadolinium-bearing fuel rods will have a higher fuel temperature than the non-gadolinium rods. The fuel performance parameters are evaluated for the gadolinium rods, therefore, because the thermal evaluations are more limiting in the gadolinium-bearing rods. This yields higher rod surface temperatures for evaluations such as LOCA analyses, which is both conservative and acceptable. Non-gadolinium-bearing fuel rods are evaluated for minimum temperatures, which are required by the transient analyses. Where limiting, this type of evaluation is also acceptable because the non-gadolinium-bearing fuel rods will be expected to have lower rod surface temperatures.

Fuel centerline temperatures are limited to those that correspond to linear heat rates no greater than 22.65 kilowatts per foot (kW/ft) for the 400V+ fuel, and 22.75 kW/ft for the 422V+ fuel. These heat rates are supported by generic analysis of the fuel design, and by the PINGP TSs and the cycle-specific reload safety analysis, which is performed in accordance with an NRC-approved methodology.

2.3.C Thermal and Hydraulic Design Conclusion

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed fuel upgrade on the thermal and hydraulic design of the core and the RCS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed fuel upgrade on the thermal and hydraulic design and demonstrated that the design (1) has been accomplished using acceptable analytical methods, (2) is equivalent to proven designs, (3) provides acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation and AOOs, and (4) is not susceptible to thermal-hydraulic instability. The NRC staff further concludes that the licensee has adequately accounted for the effects of the proposed fuel upgrade on the hydraulic loads on the core and RCS components. Based on this, the NRC staff concludes that the thermal and hydraulic design will continue to meet the requirements of GDC 10 and 12 following implementation of the proposed fuel upgrade. Therefore, the NRC staff finds the proposed fuel upgrade acceptable with respect to thermal and hydraulic design.

2.4 Transient and Accident Analyses

To support the acceptability of the requested fuel transition, the licensee evaluated the Updated Safety Analysis Report (USAR) Chapter 14 transients and accidents to determine which postulated sequences would be affected by the fuel transition. The licensee presented an analysis for each transient or accident sequence that was determined to be affected by the fuel transition request to demonstrate the acceptable performance of the 422V+ fuel assembly in the PINGP Units 1 and 2 cores.

During its acceptance review of the application, the NRC staff requested that the licensee distinguish between the evaluations that were performed for each sequence, and the select set of analyses that were presented in the LR. In its August 4, 2008, response, the licensee described the evaluation process (Reference 2).

The licensee clarified that, generically, the Westinghouse reload analysis methodology is based on the use of bounding analysis input assumptions (Reference 30). As such, the parameters can be applicable to multiple fuel designs.

Specifically for the requested fuel transition, the licensee determined which parameters important for the reload analysis would be changed as a result of the fuel transition. If a parameter changed, the licensee determined which transients and accidents modeled the changed parameter. For the potentially affected transients and accidents, the licensee considered whether the safety analysis of record modeled the affected or changed parameter in a conservative or non-conservative manner relative to the new fuel design. The resulting conclusions guided whether a particular transient was considered with merely an evaluation, or whether a re-analysis was performed.

In a general sense, the new fuel has a larger fuel-to-water heat transfer surface area, which provides initial conditions that would make DNBR transients more benign. Therefore, transients expected to be limited only by DNBR performance were generally not modeled. The licensee indicated that the transients that were reanalyzed included those sequences where the fuel geometry and/or associated pressure drop would have a significant impact on the transient responses.

In consideration of the licensee's clarification, the NRC staff accepts the licensee's approach of evaluating all transients and presenting a discussion of those that were determined to be potentially unbounded by the analysis of record. The NRC staff is reasonably assured that the licensee identified the correct transients because the evaluative method described above is based on the NRC-approved standard Westinghouse reload evaluation methodology, and because the NRC staff agrees that the process set forth would identify the limiting transients that should be re-analyzed.

The following sections present the NRC staff's evaluation of the licensee's transient and accident re-analyses performed in support of the requested fuel upgrade.

2.4.1 Non-LOCA Analyses

2.4.1.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition

The licensee reanalyzed this event to support the requested fuel transition. The NRC staff expects that the larger fuel pins would distribute the heat in the fuel pins over a large volume, which would reduce the fuel centerline temperature, but that this would also cause a different reactivity transient because the Doppler reactivity would perform differently at lower peak fuel temperatures. The NRC staff therefore agrees that this transient is appropriate for reanalysis.

2.4.1.1.A Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition Regulatory Evaluation

An uncontrolled control rod assembly withdrawal from subcritical or low power startup conditions 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 covered:

(1) The description of the causes of the transient and the transient itself

(2) The initial conditions

(3) The values of reactor parameters used in the analysis

(4) The analytical methods and computer codes used

(5) The results of the transient analyses

The NRC's acceptance criteria are based on:

(1) GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs

(2) GDC-20, insofar as it requires that the reactor protection system be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded as a result of AOOs

(3) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems.

Specific review criteria are contained in SRP Section 15.4.1 and Matrix 8 of RS-001.

2.4.1.1.B Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition Technical Evaluation

The uncontrolled rod cluster control assembly (RCCA) withdrawal from subcritical or low power startup condition is an American Nuclear Society (ANS) Condition II event that is characterized by the insertion of positive reactivity to the reactor core due to the inadvertent withdrawal of an

RCCA bank while the plant is in a subcritical or low power startup condition. The acceptance criteria for an ANS Condition II event are as follows:

- RCS pressure should remain below 110-percent of the design value.
- Fuel cladding integrity should be maintained by ensuring that the DNBR remains above its limit throughout the transient.
- A Condition II event should not generate a more serious incident without the occurrence of other independent faults.

The licensee reanalyzed this event for the 422V+ fuel transition in three steps. The core average nuclear power transient was first calculated using TWINKLE (Reference 22), followed by a fuel rod transient heat transfer calculation performed using FACTRAN (Reference 23) with average nuclear power input from TWINKLE. Finally, transient DNBR calculations are performed using VIPRE (Reference 18) using the peak core-average heat flux calculated with FACTRAN. This transient is analyzed using the STDP methodology.

The licensee used conservative input assumptions to model this accident, some of which included the use of (1) a conservatively low value for the Doppler power defect; (2) a positive, bounding isothermal temperature coefficient; (3) operation at hot zero power conditions as opposed to shutdown conditions; (4) limited availability of the reactor protection system; (5) bounding positive reactivity insertion; (6) most limiting power distributions; and (7) low initial power. These conditions combine a conservatively limiting set of plant conditions and core reactivity parameters to yield those conditions which would result in a maximized nuclear flux peak, and correspondingly limiting DNBR transient.

The licensee's results confirmed that the DNBR remained below its applicable limits during this transient. The limit below the first mixing vane grid, where the DNBR is evaluated using the W-3 correlation, was 1.428, and the minimum predicted DNBR was 1.866. Above the first mixing vane grid, the licensee used the WRB-1 correlation and obtained a minimum DNBR of 2.202, compared to a limit of 1.285. The licensee also evaluated peak fuel average and centerline temperatures, which at 1930°F and 2402°F respectively, were well within the 4746°F limit.

This NRC staff observed that, for this transient, a set of conservative assumptions intended to yield the limiting DNBR would be different from the set of conservative assumptions that would be required to yield a limiting fuel temperature. Therefore, the NRC staff requested additional information concerning the modeling assumptions used to analyze this transient.

The NRC staff requested that the licensee provide the following additional information:

1. Confirm whether the assumed isothermal temperature coefficient is bounding of that at lower assumed temperatures.

2. Explain how the effect of selecting input conditions to maximize heat flux results in a conservative hot rod fuel temperature, or how other input assumptions correct or compensate for the maximized heat flux.

In response to NRC staff Question 1, the licensee confirmed that the analysis of the Rod Withdrawal from a Subcritical Condition event, as described in Section 5.1 of the LR, bounds the same event at lower temperatures (Reference 3). Also, the licensee stated that the contribution of moderator temperature to isothermal temperature coefficient (ITC) is negligible during the initial part of the transient because the heat transfer time constant between the fuel and moderator is much longer than the neutron flux response time constant. The assumed most positive initial ITC value of +5 percent milli-rho (pcm)/°F is consistent with the maximum allowable ITC specified in TS Limiting Condition for Operation 3.1.3.a.

The licensee also noted that there is significant margin to the appropriate limit values for the Section 5.1 analysis.

In summary, the NRC staff requested the licensee to confirm whether the +5 pcm/°F ITC value could be exceeded at lower temperatures. The licensee clarified that the value assumed is the maximum allowed by TSs. The licensee's response is acceptable because the safety analysis uses parameter values that are limited by TSs requirements.

In response to Question 2, the licensee confirmed that the analysis considers the heat flux and the fuel temperature calculations in separate cases, and clarified what differences are assumed between the calculations.

Conceivably, the integral power delivered over the transient could be affected by moderator temperature effects on the fuel reactivity. The NRC staff was concerned whether a broader transient characterized by a slower nuclear power transient, delayed by a larger contribution of moderator temperature to positive reactivity for this transient, would have more adverse effects on fuel temperature. However, if one examines Figure 5.1.1-1 of the LR, one can see that the nuclear flux peaks and power turns around rather promptly, which the licensee attributes to Doppler feedback. The power descent slows marginally 2-3 seconds following the initial flux peak, which is likely a combination of moderator temperature effects and RCCA insertion. Three (3) seconds following the transient, RCCA insertion appears to dominate the reactivity level, and the nuclear power is more quickly reduced.

This demonstrates that the negative trip reactivity insertion compensates adequately for moderator temperature effects. What provides adequate conservatism for the integral power, therefore, is the assumption of what trip signal terminates the transient. In actuality, the NRC staff would expect several trips to actuate RCCA insertion prior to the analyzed trip, which would make the fuel temperature transient, as analyzed, adequately conservative with respect to the integral power effects in the fuel, because the transient would terminate more quickly than analyzed in the LR.

In a supplement to its RAI response, the licensee clarified how the transient is analyzed in three stages to yield limiting results relative to both the DNBR and the fuel temperature acceptance criteria (Reference 5). The NRC staff reviewed the licensee's response and agrees that the three-step analytic process will yield acceptably conservative results relative to both acceptance criteria.

The licensee did not present results for RCS pressure during this transient. Because the transient flux peak is so rapid, the NRC staff does not expect RCS pressure to be of concern during this transient.

The licensee's results demonstrate acceptable performance with respect to DNBR limits and fuel temperature. The NRC staff accepts these results for the following reasons: (1) the analyses were performed using NRC-approved analytic methods; (2) the analysis employed the conservative assumptions delineated above; and (3) the results meet the acceptance criteria.

The requested fuel transition is therefore acceptable with regard to the uncontrolled RCCA withdrawal from a subcritical or low-power startup condition.

2.4.1.1.C Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition Conclusion

The NRC staff has reviewed the licensee's analyses of the uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition and concludes that the licensee's analyses have adequately accounted for the changes in core design necessary for operation of the plant with the upgraded fuel design. The NRC staff also concludes that the licensee's analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure the SAFDLs are not exceeded. Based on these considerations, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 20, and 25 following implementation of the proposed fuel transition. Therefore, the NRC staff finds the proposed fuel transition acceptable with respect to the uncontrolled control rod assembly withdrawal from a subcritical or low power startup conditions.

2.4.1.2 Decrease in Reactor Coolant System (RCS) Flow

The 422V+ fuel transition would result in a reduction in the available bulk cooling volume in the core, which would cause DNBR transients to progress in a potentially more limiting fashion when compared to the same transient occurring in a full core of 400V+. Therefore, the licensee reanalyzed the complete loss of forced reactor coolant flow, because it results in a reduction in available DNBR margin as flow is lost in both coolant loops.

2.4.1.2.A Decrease in RCS 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 covered:

(1) The postulated initial core and reactor conditions,

(2) The methods of thermal and hydraulic analyses,

(3) The sequence of events,

(4) Assumed reactions of reactor systems component,

(5) The functional and operational characteristics of the reactor protection system,

(6) Operator actions, and

(7) The results of the transient analyses.

The NRC's acceptance criteria are based on:

(1) GDC 10, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs,

(2) GDC 15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, and

(3) GDC 26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

Specific review criteria are contained in SRP Section 15.3.1-2 and Matrix 8 of RS-001.

2.4.1.2.B Decrease in RCS Flow Technical Evaluation

2.4.1.2.B.1 Partial Loss of Coolant Flow

A partial loss of coolant flow may be caused by a mechanical or electrical failure in a reactor coolant pump (RCP) motor, or a fault in the power supply to the pump motor. The transient is characterized by a rapid increase in reactor coolant temperature. A partial loss of coolant flow may be terminated by either low flow sensed in 2/3 flow sensors on the reactor coolant loop, or by detection of the RCP underspeed.

The licensee used the RETRAN computer code to calculate the loop and core flow during the transient, the time of reactor trip based on RCP speed, the nuclear power transient, and the primary system pressure and temperature transients (Reference 24). The VIPRE computer code was then used to calculate the heat flux and DNBR transients based on the nuclear power and RCS flow from RETRAN (Reference 18). The event was analyzed using the RTDP assuming initial reactor power, RCS pressure, and vessel temperature were at their nominal values. Assumptions are made such that the core power was maximized during the initial part of the transient when the minimum DNBR was reached.

Acceptance criteria for this event include maintaining the DNBR above the safety analysis limit, and maintaining RCS pressure below 110-percent of the design pressure for each system.

The NRC staff reviewed the licensee's analysis results. The RCS pressure remains below 2300 pounds per square inch (psi) throughout the transient, and the DNBR maintains significant margin to its limits.

The NRC staff concludes that the licensee's analysis was performed using acceptable analytical models and the analysis was bounding for operation under uprate conditions. The NRC staff observed that the results of this transient sequence maintain significant margin to the applicable limits, and are less limiting than the results of the complete loss of coolant flow events. The NRC staff concludes that the plant will continue to meet the regulatory requirements following implementation of the proposed fuel transition. Therefore, the NRC staff finds the proposed fuel transition acceptable with respect to the partial loss of forced reactor coolant flow event.

2.4.1.2.B.2 Complete Loss of Coolant Flow

A complete loss of forced reactor coolant flow, an ANS Condition III event, may result from a simultaneous loss of electrical power supply or a reduction in power supply frequency to all RCPs. A decrease in reactor coolant flow occurring while the plant is at power could result in a degradation of core heat transfer and a subsequent increase in fuel temperature.

Accompanying fuel damage could then result if SAFDLs are exceeded during the transient. The RPS is engaged to mitigate the transient. The licensee conservatively applied ANS Condition II acceptance criteria to the analysis of this event. Thus, the licensee demonstrated that the DNBR limits were not exceeded, and pressure in the RCS and main steam system (MSS) remained below 110 percent of their respective design pressures. Specific review criteria are found in SRP Section 15.3.1-15.3.2.

The licensee analyzed this accident using the RTDP (Reference 14) along with the RETRAN computer code (Reference 24) to calculate the loop and core flows during the transient, the time of reactor trip based on the calculated RCP speeds, the nuclear power transient, and the primary system pressure and temperature transients. The VIPRE code (Reference 18) was then used to calculate the heat flux and DNBR transients based on the nuclear power and RCS temperature (enthalpy), pressure, and flow from RETRAN.

For the complete loss of flow event, the initiating event was assumed to be a loss of power to both pumps. Although the transient would progress differently if initiated by a frequency decay condition, it is the NRC staff's experience that the loss of power case is more limiting with respect to DNBR margin.

The licensee stated that RCS and MSS pressures for this event are bounded by the loss of load event, and as such, the system pressures were not evaluated for the transient. The NRC staff reviewed the PINGP USAR to confirm that there is sufficient margin to the pressure limits for the loss-of-load transient. The peak transient pressure in the RCS for the loss of load is approximately 2450 psi, and 110-percent of the design pressure (2500 psi) is 2750 psi. The licensee confirmed that the fuel transition will not adversely affect the progression of the loss of load transient. Because the licensee stated that the system pressures are bounded by the loss of load event, and the NRC staff confirmed that the RCS pressures for the loss of load event are acceptable, the NRC staff accepts the licensee's disposition of RCS pressure for the complete loss of flow transient.

With respect to DNBR margin, the licensee confirmed that the minimum DNBR was 1.466/1.470 (thimble cell/typical cell), which is above the DNBR limits of 1.415/1.415 (thimble/typical). The results of the transient analysis are therefore acceptable.

The NRC staff reviewed the licensee's analyses of the complete loss of reactor coolant flow and concluded the licensee's analyses were performed using acceptable analytical models. The NRC staff found that the licensee demonstrated that the RPS and safety systems will continue to ensure the minimum DNBR will remain above the SAL and pressure in the RCS and MSS will be maintained below 110-percent of the design pressures. Therefore, the NRC staff finds the proposed fuel transition acceptable with respect to the complete loss of reactor coolant flow.

2.4.1.2.C Decrease in RCS Flow Conclusion

The NRC staff has reviewed the licensee's analyses of the decrease in reactor coolant flow event and concludes that the licensee's analyses have adequately accounted for operation of the plant with the proposed fuel upgrade and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on these considerations, the NRC staff concludes that PINGP will continue to meet the requirements of GDC 10, 15, and 26, following implementation of the requested fuel transition. Therefore, the NRC staff finds the proposed fuel transition acceptable with respect to the decrease in reactor coolant flow event.

2.4.1.3 Reactor Coolant Pump (RCP) Locked Rotor

2.4.1.3.A Reactor Coolant Pump Locked Rotor Regulatory Evaluation

The licensee postulates either an instantaneous seizure of the rotor or break of the shaft of an RCP. 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 covered:

(1) The postulated initial and long-term core and reactor conditions,

(2) The methods of thermal and hydraulic analyses,

(3) The sequence of events,

(4) The assumed reactions of reactor system components,

(5) The functional and operational characteristics of the reactor protection system,

(6) Operator actions, and

(7) The results of the transient analyses.

The NRC's acceptance criteria are based on:

(1) GDC 27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained;

(2) GDC 28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in

damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; and

(3) GDC 31, insofar as it requires that the RCPB be designed with sufficient margin to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized.

Specific review criteria are contained in SRP Section 15.3.3-4 and Matrix 8 of RS-001.

2.4.1.3.B Reactor Coolant Pump Locked Rotor Technical Evaluation

The postulated locked rotor accident, an ANS Condition IV event, can result from an instantaneous seizure of the RCP rotor or the break of the RCP shaft. The ANS Condition IV event acceptance criteria were applied as follows:

(1) RCS pressure should be below the designated limit (2750 psi),

(2) Coolable core geometry is ensured by showing that the peak cladding temperature (PCT) and maximum oxidation level for the hot spot are below 2700°F and 16 percent by weight, respectively, and

(3) Activity release is such that the calculated doses meet 10 CFR 100 guidelines. At PINGP, this corresponds to a limiting amount of 20-percent of fuel rods in DNB.

Specific review criteria are found in SRP Section 15.3.3-4.

The licensee employed two primary computer codes to analyze this event. RETRAN (Reference 24) was used to calculate the loop and core flows during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The VIPRE code (Reference 18) was then used to evaluate the rods-in-DNB and calculate the PCT using the nuclear power and RCS temperature (enthalpy), pressure, and flow from RETRAN.

The licensee analyzed a postulated locked rotor, and noted that the consequences of the locked rotor accident are very similar to those of an RCP shaft break. The locked rotor causes a more rapid, initial reduction in the coolant flow, which results in a more challenging scenario with respect to DNB margin, peak pressure, and PCT. The RCP shaft break, however, would possibly leave the impeller free to spin in the reverse direction, which would ultimately reduce core flow when compared to the locked rotor scenario. The licensee stated that the postulated locked rotor accident represents the most limiting combination of conditions for this class of accidents. The NRC staff agrees with this approach, because the acceptance criteria for these postulated accidents are challenged early in the sequence of events, when the initial reduction in core flow caused by the locked rotor poses a more limiting scenario.

This postulated accident was analyzed twice. To evaluate peak RCS pressure and peak cladding temperature, the licensee employed conservative assumptions designed to maximize RCS pressure and cladding temperature transients using the STDP. Unlike the RTDP, which analyzes events at nominal conditions and applies uncertainties in a statistical process, the STDP applies uncertainties to initial conditions, which result in a conservative analytical approach. The RTDP approach was used to evaluate the percentage of rods in DNB for confirmation that it was less than 20-percent, consistent with the radiological analysis. The peak pressure and temperature analysis assumed initial core power, reactor coolant temperature, and pressure were at the most adverse conditions for steady-state, full-power operation, with allowances for calibration and instrument errors, whereas the DNB analysis used initial conditions consistent with the RTDP approach.

For the peak pressure and temperature analysis, the licensee assumed that the initial pressure was 2310 psi to allow for initial condition uncertainties in the pressurizer pressure measurement and control channels. Results were presented at the point of RCS maximum pressure, which was the lower plenum of the reactor pressure vessel.

The results of the analysis indicated a peak hot spot cladding temperature of 1926°F, peak zirconium-water reaction of 0.44-percent, and a peak RCS pressure of 2574 pounds per square inch atmospheric (psia). These conditions meet the acceptance criteria of 2700°F, 16-percent, and 2750 psia, respectively. The total number of rods in DNB is predicted to be less than 20-percent, which is the analytic limit for the radiological analysis.

The licensee did not evaluate fuel centerline temperatures, and the NRC staff requested additional information to demonstrate why fuel centerline temperatures do not need to be evaluated for this transient.

In its November 14, 2008, submittal (Reference 3), the licensee responded to the NRC staff's RAI by stating that the phenomenology of the locked rotor/shaft break event is such that fuel centerline temperature is generally non-limiting, and that the fuel centerline temperature transient is bounded by another USAR Chapter 14 event. Upon consideration by the NRC staff, this is both acceptable and reasonable, because a different, acceptably limiting fuel centerline temperature transient is analyzed by the licensee, the results of which are discussed in PINGP USAR Chapter 14.

The NRC staff reviewed the licensee's analyses of the locked rotor and pump shaft break events and concluded the licensee's analyses were performed using acceptable analytical models. The NRC staff concludes that the plant will continue to meet the regulatory requirements following implementation of the proposed uprate. Therefore, the NRC staff finds the proposed uprate acceptable with respect to the postulated RCP locked rotor and shaft break accidents.

2.4.1.3.C Reactor Coolant Pump Locked Rotor Conclusion

The NRC staff has reviewed the licensee's analyses of the sudden decrease in core coolant flow events and concludes that the licensee's analyses have adequately accounted for operation of the plant with the proposed fuel upgrade and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the 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 adequate core cooling will be maintained. Based on these considerations, the NRC staff concludes that PINGP will continue to meet the requirements of GDCs 27, 28, and 31 following implementation of the proposed fuel transition. Therefore, the NRC staff finds the proposed fuel transition acceptable with respect to the sudden decrease in core coolant flow events.

2.4.1.4 Chemical and Volume Control System (CVCS) Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant

Unborated water can be added to the RCS, via the 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 should stop this unplanned dilution before the shutdown margin is eliminated.

2.4.1.4.A CVCS Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant Regulatory Evaluation

The NRC staff's review covered:

(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, insofar as it requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including AOOs;

(2) GDC-15, insofar as it requires that the RCS and associated auxiliary, control, and protection systems be 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, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

Specific review criteria are contained in SRP Section 15.4.6 and other guidance provided in Matrix 8 of RS-001.

2.4.1.4.B CVCS Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant Technical Evaluation

The licensee did not analyze the boron dilution accidents. The installation of the 422V+ fuel assemblies, however, will reduce the amount of bulk coolant in the core region. This reduction was of concern to the NRC staff, because it could conceivably increase the rate of boron dilution arising from a CVCS malfunction.

In response to the NRC staff's concern, the licensee indicated that the change in total primary coolant volume due to the fuel transition would be nearly negligible (Reference 3). The NRC staff requested further comparisons to indicate that there would be no significant reduction in the boron dilution flow specifically to the core.

In Reference 5, the licensee presented a re-analysis of the boron dilution event, using the same methods and assumptions as in the current licensing basis analysis (aside from the assumed reduction in core coolant volume), and demonstrated that the change in boron dilution time associated with the fuel transition is indeed negligible, and that the results maintain significant margin to the 15-minute acceptance criterion for time for operator action to terminate the errant dilution.

The NRC staff therefore finds that the licensee acceptably determined that the change in the CVCS malfunction resulting in a boron dilution is reasonably unaffected by the requested fuel transition. Cycle-specific reload analyses in accordance with NRC-approved methodology (Reference 30) will confirm the acceptability of the boron dilution analyses on a cycle-specific basis.

2.4.1.4.C CVCS Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant Conclusion

The NRC staff has reviewed the licensee's supplemental information regarding the decrease in boron concentration in the reactor coolant due to a CVCS malfunction and concludes that the licensee's disposition has adequately accounted for operation of the plant under the proposed fuel upgrade conditions and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 10, 15, and 26 following implementation of the proposed fuel upgrade. Therefore, the NRC staff finds the proposed fuel upgrade acceptable with respect to the decrease in boron concentration in the reactor coolant due to a CVCS malfunction.

2.4.1.5 Rod Cluster Control Assembly (RCCA) Ejection

Control rod ejection accidents cause a rapid positive reactivity insertion that could, together with an adverse core power distribution, 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.

2.4.1.5.A RCCA Ejection Regulatory Evaluation

The NRC staff's review covered:

(1) Initial conditions

(2) Rod patterns and worths

(3) Scram worth as a function of time

(4) Reactivity coefficients

(5) The analytical model used for analyses

(6) Core parameters that affect the peak reactor pressure or the probability of fuel rod failure

(7) The results of the transient analyses

The NRC's acceptance criteria are based on GDC 28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to impair significantly the capability to cool the core. Specific review criteria are contained in SRP Section 15.4.8.

2.4.1.5.B RCCA Ejection Technical Evaluation

Control rod ejection accidents cause a rapid positive reactivity insertion that, together with an adverse core power distribution, could lead to localized fuel rod damage. Since the key acceptance criterion is maximum fuel stored energy, initial plant conditions are selected to maximize fuel stored energy. This event is considered at 0-percent and 100-percent power, and at beginning of cycle (BOC) and end of cycle (EOC) conditions. Since the RCCA ejection transient is a rapid transient, initial plant conditions, such as power level, pressure, flow, and temperature are not significant.

The licensee applied acceptance criteria to its analysis based on experimental testing and on conclusions drawn in WCAP-7588, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods." The analytical limit on stored energy for both previously irradiated and unirradiated fuel is 200 calories per gram (cal/g), and fuel melt must remain less than 10-percent of the pellet volume at the hot spot. Acceptance for pressure surges is based on not exceeding faulted-condition stress limits, and the licensee provided a generic disposition for this criterion. The NRC staff observes that these acceptance criteria are more rigorous than those contained in Regulatory Guide (RG) 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors."

The calculation of the RCCA ejection accidents is performed using a two-stage process. An average core calculation is performed using the TWINKLE spatial neutron kinetics code, followed by a hot spot analysis using FACTRAN.

The control rod ejection accident analyses for PINGP were performed at BOC and EOC, for both gadolinia-doped fuel and non-gadolinia-bearing fuel. The analysis for the non-gadolinia fuel at BOC, hot full power conditions yielded the largest peak enthalpy at 156.16 cal/g. The limiting peak fuel centerline temperature was predicted to be 4929°F for the BOC, hot full power case with gadolinia-bearing fuel, and the corresponding fuel melt prediction was 1.97-percent. These results, as well as those from the remaining cases that were evaluated, are within the acceptance criteria, and are hence acceptable.

As a result of a fuel failure during a test at the CABRI reactor in France in 1993, and one in 1994 at the NSRR test reactor in Japan, the NRC recognized that high burnup fuel cladding might fail during a reactivity insertion accident (RIA), such as a Rod Ejection event, at lower enthalpies than the limits currently specified in RG 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors." However, generic analyses performed by all of the reactor vendors have indicated that the fuel enthalpy during RIAs will be much lower than the RG 1.77 limits, based on their 3-dimensional (3D) neutronics calculations. For high burnup fuel, which has been burned so long that it no longer contains significant reactivity, the fuel enthalpies calculated using the 3D models are expected to be much less than 100 cal/g.

The NRC staff has concluded that, although the RG 1.77 limits may not be conservative for cladding failure, the analyses performed by the vendors, which have been confirmed by NRC-sponsored calculations, provide reasonable assurance that the effects of postulated RIAs in operating plants with fuel burnups up to 60 gigawatt days per metric ton uranium (GWD/MTU) will neither (1) result in damage to the RCPB, nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core as specified in current regulatory requirements.

A generic calculation of the pressure surge for an ejected rod worth of one dollar at beginning of life, hot full power, indicated that the peak pressure would not exceed faulted condition stress limits for the reactor pressure vessel. For PINGP operating with the 422V+ fuel, the severity of the RCCA ejection accidents does not exceed the worst-case analysis such that the generic disposition remains bounding and applicable.

Since fuel and clad limits are not exceeded, there is no danger of sudden fuel dispersal into the coolant, and since the peak pressure does not exceed the faulted condition stress limits, there is no danger of additional damage to the RCS. The analyses demonstrate that the fission product release as a result of fuel rods entering DNB is limited to less than 10 percent of the fuel rods in the core.

The NRC staff finds that the results and conclusions of the analyses performed for the control rod ejection accident are acceptable for operation following implementation of the proposed fuel upgrade.

2.4.1.5.C RCCA Ejection Conclusion

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

2.4.2 Loss-of-Coolant Accidents (LOCAs)

LOCAs are postulated accidents that would result in the loss of reactor coolant from piping breaks in the RCPB at a rate in excess of the capability of the normal reactor coolant makeup system to replenish it. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished. The reactor protection and ECCS systems are provided to mitigate these accidents.

2.4.2.1 LOCA Regulatory Evaluation

The NRC staff's review covered:

(1) The licensee's determination of break locations and break sizes,

(2) Postulated initial conditions,

(3) The sequence of events,

(4) The analytical model used for analyses,

(5) Calculations of peak cladding temperature, total oxidation of the cladding, total hydrogen generation, changes in core geometry, and long-term cooling,

(6) Functional and operational characteristics of the reactor protection and ECCS, and

(7) Operator actions.

The NRC's acceptance criteria are based on:

(1) 10 CFR 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance,

(2) Appendix K to 10 CFR Part 50, insofar as it establishes required and acceptable features of evaluation models for heat removal by the ECCS after the blowdown phase of a LOCA,

(3) GDC-4, insofar as it requires that structures, systems, and components important to safety be protected against dynamic effects associated with flow instabilities and loads such as those resulting from water hammer,

(4) GDC-27, insofar as it requires that reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained, and

(5) GDC-35, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA at a rate such that fuel clad damage that could interfere with continued effective core cooling will be prevented.

Specific review criteria are contained in SRP Sections 6.3 and 15.6.5 and Matrix 8 of RS-001.

2.4.2.2 LOCA Technical Evaluation

The specific evaluations performed by the licensee address the acceptance criteria contained in 10 CFR 50.46(b)(1) through (b)(3), and (b)(5). By demonstrating compliance with these criteria, regarding PCT, local and core-wide oxidation, and long-term cooling capability, the licensee adequately demonstrates compliance with 10 CFR 50.46(b)(4), which requires that the core remain in a coolable geometry.

2.4.2.2.A Large Break LOCA

By letter dated July 6, 2006, the licensee for PINGP Units 1 and 2 requested to implement new analytic methods to analyze large break LOCAs. Based on its detailed review of the licensee's (1) selection of analytic techniques, (2) implementation of analytic techniques, and (3) results of analysis, the NRC staff concluded that the licensee could use the referenced methods to analyze LOCAs and demonstrate compliance with the requirements of 10 CFR 50.46. Reference 25 contains the staff's evaluation; the NRC staff summarizes in the following section the methods, their implementation, and the analytic results, which are based on the same method as described in Reference 25 but account for the fuel transition.

Large break LOCA analyses supporting the proposed fuel upgrade were performed by the licensee using the NRC-approved Automated Statistical Treatment of Uncertainty Method (ASTRUM) best-estimate large break LOCA (BE-LBLOCA) methodology (Reference 26). The licensee's method was approved by the NRC for implementation at PINGP Units 1 and 2 by Amendment 179 to the PINGP Unit 1 Facility Operating License, and Amendment 169 to the PINGP Unit 2 Facility Operating License. The NRC staff reviewed the fuel transition-related analytic results to confirm that they were acceptable.

The purpose of the licensee's analysis was to demonstrate conformance with the 10 CFR 50.46 requirements using the ASTRUM method. Important input assumptions, as well as analytical models and analysis methodology for the BE-LBLOCA were provided. Analysis results were also provided, which showed that no design or regulatory limit related to the BE-LBLOCA would be exceeded at the analyzed conditions.

The licensee provided results of the PINGP Units 1 and 2 BE-LBLOCA analysis in the LR, assuming that each plant is operating at 1,683 MWt, in accordance with the statistical best estimate approach. The licensee correctly assumed a loss of offsite power. The licensee evaluated transition core effects and concluded that the ASTRUM analysis for the 422V+ core at both units bounds both the 400V+ and transition cores. The results for calculated PCTs, the maximum local cladding oxidations, and the maximum core-wide cladding oxidations are repeated in the following table, based on the limiting results obtained in Unit 1:

Parameter	PINGP Unit 1 Result	10 CFR 50.46 Limit
Limiting Break Size/Location	Double-ended Guillotine Break/Pump Discharge	N/A
Cladding Material	ZIRLO™	(Cylindrical) Zircaloy or ZIRLO™
Peak Cladding Temperature	1632°F	2200°F
Maximum Local Oxidation	0.62-percent	17.0-percent
Maximum Total Core-Wide Oxidation	0.014-percent	1.00-percent

These results demonstrate acceptable compliance with 10 CFR 50.46(b)(1) through (b)(3). The results are discussed further in Reference 1.

The NRC staff reviewed the ASTRUM methodology, and confirmed that, in addition to the limiting double-ended guillotine rupture, the method also considers slot breaks in the RCS cold leg. On this basis, the NRC staff finds the licensee's conclusion that the PINGP Units 1 and 2 PCT-limiting transient is a double-ended cold leg guillotine break acceptable, because uncertainties related to break type and size were included in the modeling approach.

The licensee evaluated the possibility of a return to criticality during the reflood stage of the LOCA and confirmed that an increase in the TS minimum boric acid concentration in the ECCS equipment would be needed. This change is acceptable with respect to the BE-LBLOCA evaluation because the increase is explicitly calculated and shown to prevent a return to criticality. This change is also evaluated for the potential for boric acid precipitation in Section 2.4.2.2.D of this SE, and the TS change request is evaluated in Section 3.2 of this SE.

Based on its review of the licensee's application of the ASTRUM BE-LBLOCA methodology, the NRC staff concluded that the Westinghouse BE LBLOCA methodology is acceptable for use for PINGP Units 1 and 2 in demonstrating compliance with the requirements of 10 CFR 50.46(b)(1) through (b)(3), operating with the requested fuel upgrade. The NRC staff's conclusion was based on the fact that the PINGP Units 1 and 2 analyses were conducted within the conditions and limitations, and supporting technical basis, of the NRC-approved Westinghouse BE LBLOCA methodology.

2.4.2.2.B Small Break LOCA

The small break LOCA (SBLOCA) includes all postulated pipe ruptures with a total cross-sectional area less than 1.0 square foot. The SBLOCAs analyzed are for those breaks beyond the capability of a single charging pump, resulting in the actuation of the ECCS. The analysis was performed to demonstrate conformance with the 10 CFR 50.46 requirements for the fuel upgrade conditions associated with PINGP, Units 1 and 2.

The licensee analyzed small break LOCAs using the NOTRUMP-EM as described in WCAP-10054-P-A, WCAP-10054-P-A, Addendum 2, and WCAP-10079-P-A (Reference 27). In consideration of the proposed fuel transition, the NRC staff reviewed the results to confirm that the licensee complies with the 10 CFR 50.46 requirements.

The licensee considered a spectrum of cold-leg break sizes that included equivalent diameters of 1.5, 2, 3, 4, 6, and 8 inches and an accumulator line break of 10.126 inches. At Unit 1, the 3-inch break was limiting with a PCT of 959°F, and at Unit 2, the 2-inch break was limiting with a PCT of 965°F. Intermediate break sizes were not considered because the evaluated spectrum demonstrated significant margin to the 2200°F limit set forth by 10 CFR 50.46. The licensee's results indicated that, for the 1.5, 6, 8, and 10.126-inch breaks, core uncovery was not expected at either unit.

The licensee acknowledged an NRC concern that top-oriented breaks could lead to longer term core uncovery due to the potential for RCP loop seal plugging. The licensee stated that such a scenario is very unlikely because it relies on a very delicate set of conditions. The licensee indicated that a more likely scenario would be core mixture level oscillations associated with the loop seals plugging with water and subsequently clearing due to the buildup of steam pressure in the core region. Finally, the licensee asserted that such oscillatory behavior has been demonstrated in experimental testing (Reference 28).

In Reference 28, the susceptibility of a particular plant configuration is considered as a function of vessel bypass resistance. The NRC staff requested that the licensee provide clarifying information to confirm that the vessel bypass resistances at PINGP were consistent with the studied plant configurations. The licensee provided this confirmation in Reference 3. Based on this confirmation, the NRC staff accepts the licensee's position regarding the non-susceptibility of the PINGP core design to stable post-LOCA core uncovery.

The licensee's limiting results (which occur in Unit 2) for the calculated PCT, the maximum local oxidation for cladding, and the core wide oxidation for cladding are summarized in the following table, along with the acceptance criteria of 10 CFR 50.46(b).

Parameter	NOTRUMP Results	10 CFR 50.46 Limits
Limiting Break Size	2-inch	N/A
Peak Clad Temperature	965°F	2200°F
Maximum Local Oxidation	0.01-percent	17.0-percent
Maximum Total Core-Wide Oxidation (All Fuel)	Negligible	1.0-percent

The NRC staff reviewed the licensee's evaluations of the ECCS during a SBLOCA, performed in accordance with the NOTRUMP SBLOCA methodology, for PINGP Units 1 and 2 operating at the proposed, fuel upgrade conditions. The NRC staff concludes that the licensee has acceptably demonstrated, at fuel upgrade conditions, compliance with 10 CFR 50.46 (b)(1) through (b)(3).

2.4.2.2.C Post-LOCA Long-Term Cooling and Sub-Criticality

To support its methods transition and proposed power uprate, the licensee performed calculations to demonstrate post-LOCA sub-criticality and acceptable long-term cooling.

Post-LOCA Sub-Criticality

The post-LOCA sub-criticality calculations were performed to demonstrate compliance, in part, with 10 CFR 50.46(b)(5), which requires a demonstration of acceptable long-term cooling capability. The sub-criticality calculation demonstrates that the core will remain sufficiently borated to preclude an inadvertent return to criticality. The licensee's calculation is performed in accordance with WCAP-8339, "Westinghouse Emergency Core Cooling System Evaluation Model – Summary," (Reference 29) and containment sump boron concentrations were used to develop a core reactivity limit that is confirmed as part of the Westinghouse Reload Safety Evaluation Methodology (Reference 30).

The licensee's subcriticality calculations employed assumptions that minimized available boron concentrations and maximized available boron dilution sources. The licensee also assumed uniform boron mixing in the sump, and the sump boron concentration was calculated as a function of pre-trip RCS conditions. The licensee calculated a post-LOCA subcriticality boron limit curve for fuel upgrade conditions, and stated that cycle-specific reload SEs will ensure that the core will remain subcritical following a LOCA. The NRC staff finds that the licensee adequately accounts for post-LOCA subcriticality following a LOCA, because the licensee has calculated a conservative post-LOCA sump boron concentration and uses NRC-approved reload methods to confirm that this concentration of boron will keep the core subcritical.

Post-LOCA Long-Term Core Cooling

The NRC staff reviewed documents during an October 1-2, 2008, audit at the Westinghouse Energy Center in support of the NRC staff's review of the requested fuel upgrade. During its audit, the NRC staff identified potentially non-conservative assumptions made regarding the PINGP capability for post-LOCA long-term core cooling. The NRC staff requested that the licensee consider modeling the post-LOCA long-term cooling using an evaluation model that is more closely in alignment with the requirements set forth for PCT calculations in Appendix K of 10 CFR 50. A similar evaluative approach is discussed in support of the AP1000 Design Certification Review (Reference 31).

The licensee's supplemental evaluation used a PINGP core design that was specific to the fuel upgrade (Reference 6). It employed the WCOBRA/TRAC thermal-hydraulic computer code and evaluated core cooling for nearly 3,000 seconds. A decay heat model was used that is in accordance with 10 CFR 50, Appendix K, and additional assumptions used to correct for empirically observed over-prediction of two-phase level swell are employed consistent with Section 21.6.4.2 of Reference 31.

This modeling approach is acceptable for several reasons. First, the calculation is being used to confirm a prior calculation, such that it is not the only indicator of acceptable long-term core cooling performance. Second, the WCOBRA/TRAC code is acceptable because the NRC staff has found it acceptable for LOCA calculations, and its effectiveness for long-term core cooling calculations has been tested experimentally in concert with the AP1000 design certification review. Finally, the licensee's approach is acceptable because it corrects for discrepancies between the code's modeling and observed test results.

The licensee's evaluation demonstrated that, although a second heatup occurs during an assumed safety injection interruption, this heatup is limited to approximately 800°F, and stable core coverage can be maintained for the long-term period following the postulated LOCA. Because the licensee used an evaluative approach that the NRC staff finds acceptable, and because the results of the evaluation indicate acceptable post-LOCA, long-term core cooling performance, the NRC staff concludes that the licensee has adequately demonstrated acceptable post-LOCA, long-term core cooling performance for fuel upgrade conditions at PINGP.

2.4.2.2.D Post-LOCA Boric Acid Precipitation

As boric acid is used at PINGP for post-LOCA reactivity hold-down, there is some degree of concern that the boric acid that is in the post-LOCA core coolant could concentrate in the core due to boil off of the salient water, which would leave boric acid behind. The licensee stated that a calculation was performed to account for the 422V+ fuel transition and confirmed the acceptability of a 7.5-hour low head safety injection initiation time following a LOCA.

Because PINGP contains an upper plenum injection system, the large break LOCA scenario is not evaluated. This is because the reactor quickly depressurizes following a large break LOCA, and low head safety injection flow to the upper plenum is established, which will flush boric acid from the core.

The NRC and the Pressurized Water Reactor Owners Group (PWROG) have communicated about the consistency of boric acid precipitation calculations and potentially non-conservative assumptions used in many of these calculations.

The NRC staff enumerated its technical concerns with boric acid precipitation calculation methods in Reference 32. These concerns, in summary, are as follows:

1. The assumed mixing volume for the calculation needs to be justified.
2. The precipitation calculation needs to take into account time-dependent variations in the available mixing volume, and the effect that coolant loop pressure losses can have on the liquid level in the reactor vessel.
3. The boric acid solubility limit assumed in the analysis must be justified.
4. The assumed decay heat, if based on a 10 CFR 50 Appendix K model, must be multiplied by 1.2.

For its review of this license amendment request, the NRC staff performed an audit of the licensee's boric acid precipitation calculations, and concluded that the concerns listed above have an acceptable disposition in the calculations. The calculations re-affirm the acceptability of a 7.5-hour time to initiate action to flush boric acid from the core. Because the calculations confirm the 7.5-hour swap over time and because they were performed in a manner that is responsive to the staff's concerns identified above, the NRC staff concludes that the licensee's boric acid precipitation calculations are acceptable.

The NRC staff requested additional information about the boric acid precipitation calculations to establish how the calculations account for a potential possibility of boric acid carryover and subsequent precipitation in a steam generator (SG).

In Reference 4, the licensee explained that boric acid concentration in the SG was not of concern because such concentration is limited due to de-entrainment in upper core structures and the pipe chase leading from the core to the SGs, and because of steam binding pressure losses that depress the two-phase level in the core. The NRC staff agrees that these phenomena would reduce boric acid carryover from entrained liquid exiting the core. Based on this consideration, and on the fact that boric acid is significantly less miscible in water vapor than in liquid water, the NRC staff accepts the licensee's disposition in this regard.

In consideration of the items discussed above, the NRC staff accepts the licensee's boric acid precipitation calculations as an adequate demonstration that the 7.5-hour swap over time provides an adequate measure to mitigate the effects of post-LOCA boric acid precipitation.

2.4.2.3 LOCA Conclusion

The NRC staff has reviewed the licensee's analyses of the LOCA events and the ECCS. The NRC staff concludes that the licensee's analyses have adequately accounted for operation of the plant under the proposed fuel upgrade conditions and that the analyses were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection system and the ECCS will continue to ensure that the peak cladding temperature, total oxidation of the cladding, total hydrogen generation, and changes in core geometry, and long-term cooling will remain within acceptable limits. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of GDCs 4, 27, and 35, and 10 CFR 50.46 following implementation of the proposed fuel upgrade. Therefore, the NRC staff finds the proposed fuel upgrade acceptable with respect to the LOCA.

2.5 Mechanical Design Evaluation

The Westinghouse 400V+ fuel design is currently in use at PINGP. Westinghouse 422V+ fuel design is a modification of the physical structure of the 400V+ fuel design. As noted previously, the 422V+ fuel design is currently in use at the KNPP, Point Beach Nuclear Plant, and R. E. Ginna Nuclear Power Plant, with good operating experience.

PERFORMANCE+ features included pre-oxidized cladding, protective bottom grids, ZIRLO™ grids, low cobalt nozzles, and a debris-mitigating bottom end plug. PERFORMANCE+ features are currently in use at PINGP with the 400V+ fuel.

The 0.422-inch OD fuel rod and associated mid-grid are features that were reviewed by the NRC and are in use at Point Beach, R.E. Ginna, and KNPP. The NRC was notified of the change back to a larger OD (0.422-inch) fuel rod with VANTAGE+ fuel during the Point Beach transition. The NRC was then later notified of a revision to the 422V+ design for R.E. Ginna, which introduced a balanced vane pattern to the mid-grid. The remaining VANTAGE+ features are already in use in the existing fuel at PINGP. For historical reference, VANTAGE+ features have been submitted to the NRC in the licensing topical report, "VANTAGE+ Fuel Assembly Reference Core Report," WCAP-12610 (Reference 9). The NRC provided acceptance for referencing of this topical report in license amendment requests. In its acceptance, the NRC concluded that WCAP-12610 provides an acceptable basis for the VANTAGE+ fuel assembly mechanical design up to a rod-average burnup level of 60 GWD/MTU. The VANTAGE+ fuel assembly design has received a generic NRC approval for lead rod burnups up to 62,000 MWD/MTU.

The 422V+ fuel assembly skeleton is similar to the 14x14 400V+ fuel and 14x14 STANDARD (STD) fuel, which has been in operation for many cycles in two-loop Westinghouse plants, except for those modifications necessary to accommodate higher burnup and those modifications necessary to accommodate the 0.422-inch OD fuel rod. Since the 422V+ fuel is intended to replace the Westinghouse Standard and Optimized fuel designs, the 422V+ exterior assembly envelope is similar in design dimensions.

The functional interface with the reactor internals is equivalent to those of the Westinghouse 400V+ fuel design for which PINGP is currently licensed to use. Also, the 422V+ fuel assembly is designed to be mechanically and hydraulically compatible with the Westinghouse OFA designs in full or transition cores, and the same functional requirements and design criteria previously established for the Westinghouse Optimized Fuel remain valid for the 422V+ fuel assembly. Those functional requirements and design criteria were approved generically for all standard fuel arrays including 14x14 arrays.

2.5.A Mechanical Design Regulatory Evaluation

As described in the PINGP USAR (Reference 33), the construction of PINGP was completed prior to the issuance of the 10 CFR 50, Appendix A, General Design Criteria on February 20, 1971. The PINGP was designed and constructed to comply with the intent of the AEC GDC as proposed on July 10, 1967. The AEC SE report acknowledged that the AEC staff assessed the plant as described in the final safety analysis report (FSAR), against the Appendix A Design Criteria and found that the plant design generally conforms to the intent of these Criteria.

Although PINGP is not a GDC or SRP plant, GDC 16, GDC 38, GDC 50 and SRP Sections 4.2, 4.3, 4.4, 6.2.1.3, 6.3, and 15 can be used to provide guidance for the review of the mechanical design aspects.

2.5.B Mechanical Design Technical Evaluation

There are precedents for the use of the 422V+ fuel. The 422V+ fuel design was previously reviewed and approved by the NRC and is currently in use at the R. E. Ginna, Kewaunee, and Point Beach nuclear plants. The current operating experience at Ginna, Kewaunee, and Point Beach supports the use of the 422V+ fuel design at PINGP. The 400V+ fuel currently installed at PINGP Units 1 and 2 will be replaced with 422V+ fuel. Both fuel types are manufactured by Westinghouse Nuclear. The mechanical design features are described in Section 2 of Attachment 4 of Enclosure 1 of the licensee's submittal (Reference 1). The comparison between 400V+ and 422V+ fuel is fully described in Section 2 of the LR in the licensee's application.

As approved in WCAP-12610, VANTAGE+ fuel uses ZIRLO™ cladding and the dimensional modifications to the fuel assembly skeleton were introduced for extended burnup applications. The primary feature of the fuel change is a change in diameter from 0.400 to 0.422 inches.

The significant new mechanical features of the 422V+ design for PINGP include a 0.422-inch OD fuel rod and a 0.422-inch OD instrumentation tube. The new 400V+ style (2.25-inch tall, vertical springs, horizontal dimples) mid-grid, is designed to be compatible with the 0.422-inch OD fuel rods and 14x14 400V+ mixed fuel cores.

The significant new mechanical features of the 422V+ design in comparison to the current 400V+ design include: (1) 0.422-inch outer diameter (OD) fuel rod, (2) 0.422-inch OD instrumentation tube, and, (3) a new 400V+ style (2.25-inch tall, vertical springs, horizontal dimples) ZIRLO™ mid-grid; designed to be compatible with the 0.422-inch OD fuel rods as well as 14x14 400V+ mixed fuel cores. The ZIRLO™ 422V+ mid-grid design combines the enhanced anti-snag geometry and reduced pressure drop performance in the 400V+ style package. This mid-grid design evolution started with the original STD Inconel mid-grid. This design was modified to an OFA style Zircaloy vane-less design that has been used extensively at the Zorita Nuclear Power Plant in Spain and at Point Beach. From the Zorita design, the mid-grid was adopted for the Point Beach and KNPP by adding mixing vanes, changing the material composition from Zircaloy-4 to ZIRLO™, and other minor design changes.

Since the 422V+ fuel is intended to replace the Westinghouse Standard and Optimized fuel designs, the 422V+ exterior assembly envelope is similar in design dimensions. The functional interface with the reactor internals is equivalent to those of the Westinghouse 400V+ fuel design for which PINGP is currently licensed. Also, the 422V+ fuel assembly is designed to be mechanically and hydraulically compatible with the Westinghouse OFA designs in full or transition cores, and the same functional requirements and design criteria previously established for the Westinghouse Optimized Fuel remain valid for the 422V+ fuel assembly. Those functional requirements and design criteria were approved generically for all standard fuel arrays including 14x14 arrays. Table 2-1 of the LR in Reference 1 compares the 422V+ fuel assembly to the current Westinghouse 400V+ fuel design.

Based on the NRC staff's review of the mechanical analyses and evaluations of the effects of the 422V+ and 400V+ design differences as described in Sections 2 and 6 of Attachment 4 of Enclosure 1 of the licensee's submittal, it is concluded that the two designs are mechanically compatible with each other. The 422V+ fuel rod mechanical design bases remain relatively unchanged from those used for the 400V+ assemblies as currently operating in PINGP Cycle 25, other than the change in the fuel rod radial dimension (that is, 0.422-inch OD versus 0.400-inch OD).

The criteria pertinent to the fuel rod design are rod internal pressure, cladding stress and strain, cladding oxidation and hydriding, fuel temperature, cladding fatigue, cladding flattening, fuel rod axial growth, plenum cladding support, cladding free-standing, and end-plug weld integrity. Each of these key fuel rod design criteria was evaluated by the licensee for use with the Westinghouse 422V+ fuel assembly design for PINGP. Based on these evaluations, the NRC staff concludes that each design criteria can be satisfied through transition cycles to a full core of the 422V+ design.

In response to the NRC staff's request for additional information (RAI) (References 34 and 35) regarding the specific section, subsection, and edition of the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME Code) utilized for the cladding stress limits, the licensee clarified and corrected their initial response (Reference 34) and stated that the Code used is ASME Code Section III, Article NG-3000, 1998 Edition. The NRC staff finds this acceptable because it is consistent with the SRP Section 4.2 recommendations, and is also in agreement with the NRC-approved licensing basis documentation.

In its response to the NRC staff's RAI regarding the calculated cladding cumulative usage factor (CUF), the licensee stated that there is a reduction in CUF for the 422V+ fuel reload transition. In response to the NRC staff's RAI on end plug weld tensile pressure differential loads and allowable limits, the licensee stated that the fuel rod weld plug integrity analyses were performed on a generic basis and concluded that the fuel system will not be damaged due to excessive end plug weld tensile pressure differential loads. The licensee also concluded that the 422V+ fuel rod design has increased margin due to the lower EOL rod internal pressure compared to that of the 400V+ fuel rod design. The NRC staff finds this acceptable because of the increased margin associated with the 422V+ fuel rod design as compared to the 400V+ fuel rod design.

In response to the NRC staff's RAI to provide a summary of the computed stresses and the corresponding allowable limits for fuel rods and thimble tubes for the 422V+ fuel assembly design, the licensee provided a table of generic stress results and the corresponding allowable limits. The NRC staff reviewed these results and finds them acceptable as there is substantial margin available for Operating Basis Earthquake (OBE) loading as well as for combined loading from LOCA, and Safe Shutdown Earthquake (SSE).

The NRC staff requested the licensee to provide the maximum grid impact force from SSE and LOCA analyses. In its response, the licensee provided a table of grid impact forces along with the associated limits. The grid loads are less than the allowable limits except for the 3 fuel assembly arrays for SSE/LOCA loading. The licensee performed an evaluation for those 3 fuel assembly arrays that demonstrated that the core coolable geometry is satisfied in the presence of the few crushed grids for 422V+ as well as 400V+ fuel assemblies. For OBE loading, no grid crush was predicted. This is acceptable to the NRC staff because coolable geometry is maintained for SSE/LOCA loading in the presence of those grids that were postulated to crush, due to excessive grid impact force.

The NRC staff requested a summary of stresses in the fuel rods and thimble tubes of the 422V+ fuel assembly under seismic and LOCA loadings. In its RAI response, the licensee submitted a table of the requested stresses from a generic bounding analysis. The NRC staff finds those reported stresses acceptable because the generic results bound the values for PINGP, and because there is substantial margin that exists between the generic results and the allowable limits.

In response to the NRC staff's request, the licensee provided a summary of results of the maximum combined LOCA and SSE loads along with the allowable load for the 14x14 type guide tube. The NRC staff reviewed the load summary and found it to be acceptable because the applied load is less than the allowable load, thus maintaining the structural integrity of the guide tube.

Upon the NRC staff's request, the licensee provided a summary of the flow induced vibration alternating stress levels along with the corresponding ASME Code endurance limit for high cycle fatigue for core barrel components, thermal shield components, and the lower core support plate. The licensee summarized the alternating stress levels and the corresponding high cycle endurance limit in the thermal shield support due to the RCP induced vibration. The licensee provided to the NRC staff the flow induced vibration forces in the upper support column base and top welds compared with the corresponding force in the reference 2-loop plant. In addition, the licensee provided the NRC staff with a summary table of the computed primary plus secondary stresses along with the allowable stresses and the CUF values for the reactor vessel internals. The NRC staff reviewed these forces, stresses, and CUFs and finds them acceptable because they are below the corresponding allowables, thus assuring the structural integrity of the reactor vessel internals for the 422V+ fuel assembly design.

Based on the evaluation of the 422V+ and 400V+ design differences, the licensee has adequately demonstrated that the two designs are mechanically compatible with each other. The 422V+ fuel rod mechanical design bases remain relatively unchanged from those used for the 400V+ assemblies currently operating at PINGP Cycle 25, other than the change in the fuel rod radial dimension (that is, 0.422-inch OD versus 0.400-inch OD). In addition, the 422V+ fuel rod mechanical design is similar to those in operation at Point Beach Units 1and 2 and KNPP. Evaluations and analyses were performed by Westinghouse for the licensee, which confirmed the acceptable use of these features for PINGP operations.

The NRC staff finds that the fuel assemblies are structurally designed to perform satisfactorily throughout their lifetime. The loads, stresses, and strains resulting from the combined effects of flow induced vibrations, earthquakes, reactor pressure, fission gas pressure, fuel growth, thermal strain, and differential expansion during both steady state and transient reactor operating conditions have been adequately considered in the design of the fuel rods and fuel assembly.

The NRC staff has reviewed and agrees with the licensee's evaluations which demonstrate that there is no impact of having fuel with more than one type of geometry simultaneously existing in the core during the transition cycles from a structural adequacy considerations standpoint.

2.5.C Mechanical Design Conclusion

The NRC staff has reviewed the licensee's supporting technical information for the proposed license amendment request pertaining to fuel change. Based on the above evaluation, the NRC staff concludes that the mechanical design aspect of the fuel change to Westinghouse 422V+ is acceptable.

2.6 Containment Systems Evaluation

The NRC staff has reviewed the proposed amendments to the LOCA Containment Peak Pressure Response analyses and the Steam Line Break Inside Containment analyses. The following is an evaluation for these analyses.

2.6.A Containment Systems Regulatory Evaluation

The licensee addressed the regulatory requirements applicable to the proposed amendment in Section 4.1 of Enclosure 1 of the application dated June 26, 2008. As described in this enclosure and confirmed by the PINGP USAR, Section 1.2 "Principal Design Criteria", PINGP was designed and constructed to meet the intent of the AEC GDC for Nuclear Power Plant Construction Permits, as originally proposed on July 1967. Plant construction was significantly completed prior to the issuance of the February 20, 1971, 10 CFR Part 50, Appendix A GDC. However, the AEC Safety Evaluation Report acknowledged that the AEC staff assessed the plant, as described in the FSAR, against the Appendix A design criteria and "... are satisfied that the plant design generally conforms to the intent of these criteria."

The regulatory requirements, criteria, and guidance applied by the NRC staff in the review of the proposed changes are as follows:

- Criterion 16, "Containment design," which establishes in part, that reactor containment and associated systems shall be provided to assure that the containment design conditions important to safety are not exceeded for as long as postulated accidents require.

- Criterion 38, "Containment heat removal," which establishes in part, that a system to remove heat from the reactor containment shall be provided to reduce rapidly, the containment pressure and temperature following any LOCA and maintain them at acceptably low levels.

The PINGP licensing basis for the containment is described in the licensee's USAR Sections 5.2.1.1 and 14.9.3. PINGP was also licensed prior to the issuance of the NRC SRP. However, the PINGP licensing basis provides acceptance criteria equivalent to the NRC SRP criteria. As is standard practice in such cases, the NRC staff has used the SRP, as well as the applicable Westinghouse topical reports as guidance in reviewing the licensee's analyses supporting the fuel transition.

The NRC staff review also considered the relevant information contained in the following documents:

- SRP Section 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents," contains guidance on the review of the analysis of the mass and energy release to assure that the data used to evaluate the containment and subcompartment functional design are acceptable.

- SRP Section 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures," and Section 6.2.2, "Containment Heat Removal Systems," contains guidance on the review of pressurized-water reactor (PWR) main steam line break accidents.

- WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design March 1979 Model," dated May 1983.
- WCAP-8822, "Mass and Energy Releases Following a Steam Line Rupture," dated September 1976.
- WCAP-16219-P, "Development and Qualification of a GOTHIC Containment Evaluation Model for the PINGP," dated April 2004.

2.6.B Containment Systems Technical Evaluation

2.6.B.1 LOCA Mass and Energy Releases Inside Containment

2.6.B.1.1 Overview

The licensee's analysis postulates a LOCA and containment pressurization that must be considered for the PINGP Units 1 and 2 with 422V+ fuel at fuel upgrade (FU) operating conditions. The uncontrolled release of pressurized high-temperature reactor coolant from the primary system piping, known as LOCA, will inject high-energy steam and water into the containment. This release of mass and energy to the containment will increase containment pressure and temperature.

2.6.B.1.2 Mass and Energy Releases Inside Containment

The long-term LOCA mass and energy releases for the hypothetical double-ended pump suction (DEPS) break with minimum safeguards, the DEPS break with maximum safeguards, and double-ended hot-leg (DEHL) rupture break cases represent the bounding containment pressurization accident sequences for the large break loss of coolant events.

The Westinghouse evaluation model used for the long-term LOCA mass and energy release calculations is the March 1979 model described in WCAP-10325-P-A. The WCAP-10325-P-A mass and energy release evaluation model is comprised of mass and energy versions of the following codes: SATAN-VI, WREFLOOD, FROTH, and EPITOME. Theses codes were used to calculate the long-term LOCA mass and energy releases for PINGP. All input parameters and assumptions are chosen consistent with the accepted analysis methodology found in WCAP-10325-P-A. These are listed in Tables 5.3-1 through 5.3-3 of Enclosure 1, Attachment 4, of the licensee's June 26, 2008, application (Reference 1). One mass and energy release model was created that bounds both PINGP units.

The parameters pertaining to the WCAP-10325-P-A methodology were reviewed for differences between the analyzed conditions and the FU conditions. The reactor power including uncertainties, 1,683 MWt, used in the calculation of the mass and energy releases is the same as the reactor power including uncertainties for the FU analyzed conditions. The use of plant-specific core power uncertainty in the WCAP-10325-P-A methodology was approved by the NRC. The bounding core-stored energy value of 5.77 full-power seconds (FPSs) used in the mass and energy release calculation is a conservative value that bounds the core stored energy of the 422V+ fuel product (4.60 FPS) at the FU analyzed conditions.

The enthalpy of the SG secondary system steam/water is essentially unchanged at the FU analyzed conditions and the mass of the secondary system steam/water is slightly decreased at the higher output power level, consisting of the reactor power including uncertainties. Therefore, the sensible energy of the SG secondary system that can be released to the containment is essentially unchanged due to the FU analyzed conditions.

Based on the evaluation above, the NRC staff has determined that the LOCA containment mass and energy releases calculated for PINGP Units 1 and 2 are conservative and applicable to the 422V+ fuel product at the FU analyzed conditions.

The mass and energy analysis for the DEHL break, the DEPS break with minimum safeguards, and the DEPS break with maximum safeguards were provided for use in the containment peak pressure integrity analysis below for a limiting case.

2.6.B.1.3 Containment Peak Pressure Response Analysis

A break in the primary RCS piping causes a loss of coolant, which results in a rapid release of mass and energy to the containment atmosphere. The blowdown phase causes a rapid increase in the containment pressure, which results in the actuation of the emergency fan cooler and containment spray systems.

The RCS accumulators begin to refill the lower plenum and downcomer of the reactor vessel with water after the end of blowdown. The reflood phase begins after the vessel fluid level reaches the bottom of the fuel. During this phase, the core is quenched with water from both the accumulators and pumped safety injection. The quenching process creates a large amount of steam and entrained water that is released to containment through the break.

The LOCA mass and energy release decreases with time as the system cools. Core decay heat is removed by nucleate boiling after the reflood and froth phases are complete. The long term core fluid level is maintained by pumping water back into the vessel from the sump recirculation system. The containment heat removal systems continue to condense steam and reduce the containment pressure and temperature over time.

The containment integrity analysis uses the GOTHIC version 7.1 patch 1 computer codes to calculate the peak pressure and temperatures for the mass and energy releases. The PINGP GOTHIC containment evaluation model consists of a single lumped-parameter node and the diffusion layer model (DLM) is used for heat transfer to all structures in the containment. The model was described in WCAP-16219 and the evaluation model was approved by the NRC in a letter dated August 2005. The containment analysis methodology satisfies the current NRC acceptance criteria from 10 CFR 50, Appendix A and applicable SRP sections.

The containment pressure, gas temperature, liner temperature and water (sump) temperature transients from each of the LOCA cases are shown in Figure 5.3-1 through Figure 5.3-4 of the LR contained in Reference 1. In this same application, Table 5.3-8 summarizes the peak LOCA containment response results for the three cases studied.

For the DEHL case, it was determined that the peak pressure was 42.9 per square inch gauge (psig) at 18.0 seconds. For the DEPS with minimum safeguards, the containment pressure reached a peak of 43.5 psig at 3,601 seconds. For the DEPS with maximum safeguards, the peak pressure was determined to be 40.6 psig at 14 seconds.

According to the licensee's analysis results, the peak containment pressure was less than the containment design pressure of 46 psig for all three cases, the containment pressure was less than 50 percent of the peak value within 24 hours and the containment liner temperature was less than 268°F in all three cases. Based on these results, the NRC staff has determined that all applicable containment integrity acceptance criteria for PINGP have been met. Therefore, the NRC staff finds that the bounding LOCA containment response analysis for PINGP with 422V+ at FU analyzed operating conditions is acceptable.

2.6.B.2 Steam Line Break Inside Containment

2.6.B.2.1 Overview

The licensee's analysis postulates a steam line break inside containment for the 422V+ Fuel Transition Program. Steam line ruptures occurring inside a reactor containment structure may result in significant releases of high-energy fluid to the containment environment that could produce high-pressure conditions for extended periods of time. There are competing effects for variations in the postulated accident scenario, which makes it difficult to determine the worst cases for containment pressure following a steam line break. Therefore, the steam line break and containment response analysis considers a spectrum of cases that vary the break size, initial power condition, and the postulated single failure.

The GOTHIC containment model and input assumptions for PINGP are documented in WCAP-16219-P and the initial conditions that are used in this containment analysis are summarized in Table 5.4-1 of the LR.

Separate analyses have been done for Unit 1 and Unit 2 due to the different SG models.

2.6.B.2.2 Mass and Energy Releases Inside Containment

The Westinghouse steam line break mass and energy release methodology is documented in WCAP-8822, "Mass and Energy Releases Following a Steam Line Rupture", which was approved by the NRC in an August 12, 1983 letter. WCAP-8822 forms the basis for the assumptions used in the calculation of the mass and energy releases resulting from a steam line rupture. The licensee used the Westinghouse LOFTRAN computer code to calculate the mass and energy release into the containment as a result of the postulated rupture of a steam line. The LOFTRAN computer code has been previously approved by the NRC for steam line break analyses.

2.6.B.2.3 Containment Peak Pressure Response Analysis

The containment integrity analysis uses the GOTHIC computer code. The PINGP GOTHIC containment evaluation model consists of a single lumped-parameter node; the DLM is used for heat transfer to all structures in the containment, which was described in WCAP-16219-P and approved by the NRC in a letter dated August 12, 2005 (ADAMS Accession No. ML052000046). The model and sample analyses in WCAP-16219-P used Version 7.1 patch 1 of GOTHIC 7(QA). However, the current analysis has used the most recent release of GOTHIC, Version 7.2a, to ensure that any corrections to coding errors are addressed. No new user options are implemented and the input model is consistent with what has been previously approved by the NRC.

The licensee analyzed a spectrum of cases which varied the break size, the initial power level, and the single failure, in order to encompass the many factors that influence the quantity and rate of the mass and energy release from the steam line. The specific initial power levels that are analyzed are 100 percent, 70 percent, 30 percent, and 0 percent as presented in WCAP-8822.

The largest possible break is a double-ended rupture (DER) immediately downstream of the SG outlet. The DER credits liquid entrainment in the initial blowdown phase. A smaller break size without crediting any liquid entrainment is also analyzed.

In its June 26, 2008, application, the licensee states that the entrainment characteristics for large steam line breaks are not sensitive to the SG design. In a letter dated March 12, 2009 (ADAMS Accession No. ML090721087), in response to the NRC staff's RAI, the licensee elaborated. The licensee points out that a comparison was made between the Framatome Model 56/19 SG in PINGP Unit 1, to the Westinghouse Model 51 SG that is in PINGP Unit 2, and for which the break quality results from WCAP-8822 have been applied. The Framatome SG was designed as a direct replacement to the model 51 SG and is therefore very similar in size and capacity. The largest difference is the moisture separator. The Framatome Model 56/19 SG has 16 swirl vane separators while the Westinghouse Model 51 SG has only 3. The conclusion from the review of the Framatome Model 56/19 design compared to the Westinghouse Model 51 design is that they are very similar. Differences in the separator design are determined not to be important since the two SG designs used in the TRANFLO calculations in WCAP-8821 and WCAP-8822 were also diverse. These differences are also determined to not be important relative to the liquid released from a large steam line break.

The single failures that have been postulated for PINGP that would impair the performance of various steam line break protection systems include a containment safeguards failure, an auxiliary feedwater pump runout protection failure and a feedwater relief valve failure. The licensee's analysis cases separately consider each single failure at each initial power.

According to the licensee's analysis results, the limiting containment pressure case is the largest DER initiated from 30-percent power with a containment safeguards failure. For Unit 1, the limiting case is a 1.4 ft^2 break, resulting in a peak containment pressure of 44.2 psig. For Unit 2, the limiting case is a 4.6 ft^2 break, resulting in a peak containment pressure of 45.9 psig. This analysis has demonstrated that the containment pressure remains below the containment design pressure throughout the transient for a postulated secondary system pipe rupture.

Based on these results, the NRC staff has determined that all applicable containment integrity acceptance criteria for PINGP have been met. Therefore, the NRC staff finds that the postulated steam line break inside containment analyses for PINGP with 422V+ fuel at FU analyzed operating conditions are acceptable.

2.6.C Containment Systems Conclusions

Based on its review of the licensee's analysis, the NRC staff finds that the proposed analyses discussed in this section of the SE are acceptable in support of the transition from Westinghouse 400V+ fuel to the 422V+ fuel.

3.0 CHANGES TO TECHNICAL SPECIFICATIONS

3.1 Section 2.1.1, Reactor Core SLs

The licensee will revise the peak fuel centerline temperature for UO2 and Gd_2O_3-UO2 fuel design to reflect the use of burnable absorbers. TS 2.1.1.2 will be revised as follows:

The peak fuel centerline temperature shall be maintained as follows:

(a) < 5080 °F, decreasing by 58 °F per 10,000 MWD/MTU burnup, for fuel containing UO2;

(b) < (5080 °F minus 6.75 °F per w/o Gd_2O_3), decreasing by 58 °F per 10,000 MWD/MTU burnup, for fuel containing gadolinia.

The revised fuel centerline temperature for UO2 and Gd_2O_3-UO2 fuel design is a standard form and has been accepted by the NRC staff. The NRC staff requested that the licensee clarify the basis for the requested change, and the licensee indicated the following (Reference 4):

- The requested Technical Specifications change is based on avoiding the fuel melting temperatures for both uranium dioxide fuel and gadolinia-doped fuel. The technical basis is contained in NRC-approved, generic licensing topical reports for Westinghouse fuel designs.

- The requested Technical Specifications change is based on fuel material properties that are independent of fuel geometry and therefore, the change is justified for and applicable to the co-resident fuel assemblies in the PINGP core designs.

The NRC staff observes that the technical basis for this requested change is contained in NRC-approved topical reports describing the Westinghouse uranium dioxide fuel designs, and has been incorporated into the Standard Technical Specifications (STSs). On the basis that the NRC has generically approved the technical basis for the peak fuel centerline temperatures, and because the requested change is consistent with the applicable STSs, the NRC staff finds the proposed revision to TS 2.1.1.2 acceptable. The appropriate topical reports are listed in Reference 4.

3.2 Section 3.5.1.4, Increase Accumulator Boric Acid Concentration

The licensee requested to increase the minimum boron concentration in the safety injection accumulators from 1900 ppm to 2300 ppm. The licensee indicated that the change was necessary to assure the necessary quality of the safety injection system to perform its intended safety function of holding the core subcritical following a design basis LOCA.

Safety injection accumulator boric acid concentration is constrained by, among others, two primary parameters. At a minimum, the boric acid concentration must be sufficiently high that it can be shown to hold the core subcritical following a LOCA in its condition of greatest excess reactivity. At a maximum, the boric acid concentration must not be so high that it would cause the precipitation of boric acid in the post-LOCA core region that could inhibit long-term core cooling.

The minimum concentration is confirmed to be acceptable via the NRC-approved Westinghouse reload SE process described in Reference 29. Therefore, the NRC staff finds the requested increase acceptable because it is confirmed to provide adequate boron to ensure that the accumulators can perform their intended safety functions on a cycle-specific basis using NRC-approved methods. Should the minimum concentration be shown to be ineffective, the licensee would be required to submit another license amendment request.

The accumulator maximum concentration assumed in the boric acid precipitation calculation is unchanged at 3500 ppm, and is acceptable as described in Section 2.4.2.2.D of this SE.

3.3 Changes to TS Section 4.3 – Design Features, Fuel Storage

The licensee proposed the following changes to TS Section 4.3:

a. TS 4.3.1.1 b, Fuel Storage – Criticality

   Change "Reference 1" to read "USAR Section 10.2."

b. TS 4.3.1.1 c, Fuel Storage – Criticality

   Change "Reference 1" to read "USAR Section 10.2."

c. TS 4.3.1.2 b, Fuel Storage – Criticality

   Change "Reference 2" to read "USAR Section 10.2."

d. TS 4.3.3, Fuel Storage – Capacity

   Change "Ref. 3" to read "USAR Section 10.2."

e. TS 4.3, References

   Delete References 1, 2 and 3.

3.3.1 Regulatory Evaluation of Changes to TS Section 4.3

The regulations in 10 CFR Part 50.36 specify the content of TSs. Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.36(c)(4), Design Features, requires features to be included that are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety, and which are not covered in the other TS categories described in 10 CFR 50.36(c).

The NRC staff reviewed the proposed changes for compliance with 10 CFR 50.36 and agreement with the precedent as established in NUREG-1431. In general, licensees cannot justify TS changes solely on the basis of adopting the model, Standard Technical Specifications Westinghouse Plants STS format and content. To ensure this, the NRC staff makes a determination that proposed changes maintain adequate safety. Changes that result in relaxation (less restrictive condition) of current TS requirements require detailed justification.

In general, there are two classes of changes to TSs: (1) changes needed to reflect contents of the design basis (TSs are derived from the design basis), and (2) voluntary changes to take advantage of the evolution in policy and guidance as to the required content and preferred format of TS over time. This amendment to Section 4.3 deals with the second class of change; namely, changes that take advantage of the preferred format of TSs, which in this application will result in establishing the change control requirements of 10 CFR 50.59 as the appropriate level of regulatory control for fuel storage design features.

Licensees may revise the TSs to adopt NUREG-1431 format and content provided that plant-specific review supports a finding of continued adequate safety because: (1) the change is editorial, administrative, or provides clarification (i.e., no requirements are materially altered); (2) the change is more restrictive than the licensee's current requirement; or (3) the change is less restrictive than the licensee's current requirement, but nonetheless still affords adequate assurance of safety when judged against current regulatory standards. The detailed application of this general framework, and additional specialized guidance, are discussed in Section 3.3.2 in the context of the specific proposed changes.

3.3.2 Technical Evaluation of Changes to TS Section 4.3

PINGP TS Section 4.0, Design Features, contains design information regarding the plant site (TS 4.1), the reactor core (TS 4.2), and fuel storage (TS 4.3). Some of this information is also contained in the USAR. Additionally, some of the information does not conform to the format and content provided in NUREG-1431, Standard Technical Specifications (STS), Westinghouse Plants. The licensee proposes to change TS Section 4.3 to remove explicit references to methodologies and descriptions contained in the USAR, and align the associated TS information to conform to NUREG-1431. The deleted TS information is contained in the USAR. The USAR information is subject to the change control requirements of 10 CFR 50.59, Changes, Tests and Experiments, which would include any future changes to the information removed from TS.

In its application, the licensee provided the following information in support of the proposed TS changes. The NRC staff has performed an evaluation of the requested changes for compliance with 10 CFR 50.36 and the content and format identified in NUREG-1431, as discussed below.

3.3.3 NRC Staff Evaluation of Proposed Changes to TS 4.3, Fuel Storage

a. TS 4.3.1.1 b, Fuel Storage – Criticality

The licensee's application described the change requested in TS 4.3.1.1 b as follows:

> The change to delete Reference 1 is justified by referring directly to Section 10.2 of the USAR, which contains this information. Reference 1 in Specification 4.3.1.1 b is Reference 45 of the USAR. This change is consistent with the Fuel Storage Design Features Specification in NUREG-1431, Revision 3.

TS 4.3.1.1 b makes explicit reference in the TS to the "Prairie Island Units 1 and 2 Spent Fuel Pool Criticality Analysis", WCAP-16517 NP, Revision 0, Westinghouse Electric Company, November 2005, when discussing allowances for uncertainties in TSs. The NUREG-1431 STS do not explicitly reference the specific criticality analyses, but instead reference the section in the USAR where the allowance for the uncertainties would typically be described. This change removes information that is contained in the USAR to adopt the preferred format of TSs, which for this change will result in establishing the change control requirements of 10 CFR 50.59 as the appropriate level of regulatory control for fuel storage design features. No regulatory requirements are materially or otherwise altered by the proposed TS change. The proposed change is administrative and editorial in nature and is consistent with the requirements of 10 CFR 50.36 and with the content and format of NUREG-1431. Therefore, the NRC staff finds the proposed change to PINGP TS Section 4.3.1.1 b acceptable.

b. TS 4.3.1.1 c, Fuel Storage – Criticality

The licensee's application described the change requested in TS 4.3.1.1 c as follows:

> The change to delete Reference 1 is justified by referring directly to Section 10.2 of the USAR, which contains this information. Reference 1 in Specification 4.3.1.1 c is Reference 45 of the USAR. This change is consistent with the Fuel Storage Design Features Specification in NUREG-1431, Revision 3, and the Fuel Storage Design Features Specifications in plant-specific TS for plants such as: Beaver Valley Units 1 and 2, Callaway, Point Beach, and various other Westinghouse plants.

TS 4.3.1.1 c makes explicit reference in the TS to the "Prairie Island Units 1 and 2 Spent Fuel Pool Criticality Analysis", WCAP-16517 NP, Revision 0, Westinghouse Electric Company, November 2005, when discussing allowances for uncertainties in TSs. The improved STS do not explicitly reference the specific criticality methodology, but instead reference the section in the USAR where the allowance for the uncertainties would typically be described. This change removes information that is contained in the USAR to adopt the preferred format of technical specifications, which for this change will result in establishing the change control requirements of 10 CFR 50.59 as the appropriate level of regulatory control for fuel storage design features. No regulatory requirements are materially or otherwise altered by the proposed TS change. The proposed change is administrative and editorial in nature and is consistent with the requirements of 10 CFR 50.36 and with the content and format of NUREG-1431. Therefore, the NRC staff finds the proposed change to PINGP TS Section 4.3.1.1 c acceptable.

c. TS 4.3.1.2 b, Fuel Storage – Criticality

The licensee's application described the change requested in TS 4.3.1.2 b as follows:

> The change to delete Reference 2 is justified by referring directly to Section 10.2 of the USAR, which contains this information. Reference 2 in Specification 4.3.1.2 b is Reference 39 of the USAR. This change is consistent with the Fuel Storage Design Features Specification in NUREG-1431, Revision 3, and the Fuel Storage Design Features Specifications in plant-specific TS for plants such as: Beaver Valley Units 1 and 2, Callaway, Point Beach, and various other Westinghouse plants.

TS 4.3.1.2 b makes explicit reference in the TS to the "Criticality Analysis of the Prairie Island Units 1 and 2 Fresh and Spent Fuel Racks", Westinghouse Commercial Nuclear Fuel Division, February 1993, when discussing allowances for uncertainties in this TS. The improved STS do not explicitly reference the specific criticality methodology, but instead reference the section in the USAR where the allowance for the uncertainties is described. This change removes information that is contained in the USAR to adopt the preferred format of technical specifications, which for this change will result in establishing the change control requirements of 10 CFR 50.59 as the appropriate level of regulatory control for fuel storage design features. No regulatory requirements are materially or otherwise altered by the proposed TS change. The proposed change is administrative and editorial in nature and is consistent with the requirements of 10 CFR 50.36 and with the content and format of NUREG-1431. Therefore, the NRC staff finds the proposed change to PINGP TS Section 4.3.1.2 b acceptable.

d. TS 4.3.3, Fuel Storage – Capacity

The licensee's application described the change requested in TS 4.3.3 as follows:

> The change to delete Reference 3 is justified by referring directly to Section 10.2 of the USAR, which contains this information.

TS 4.3.3 refers to Reference 3 of TS 4.0, which is "USAR, Section 10.2". The proposed change deletes "Ref. 3" and refers directly to "USAR Section 10.2" in the text of the TS. No regulatory requirements are materially or otherwise altered by the proposed TS change. The proposed change is administrative and editorial in nature and is consistent with the requirements of 10 CFR 50.36. Therefore, the NRC staff finds the proposed change to PINGP TS Section 4.3.3 acceptable.

e. TS 4.3, References

The licensee's application described the change requested in TS 4.3 as follows:

> The change to delete References 1, 2 and 3 is conforming to the above changes and is justified by referring directly to Section 10.2 of the USAR in the text of the TS.

The proposed conforming change is administrative in nature and is consistent with the requirements of 10 CFR 50.36. Therefore, the NRC staff finds the proposed change acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes the surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (73 FR 54866). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the

Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Wadley, Michael D., Nuclear Management Company, letter to U.S. NRC, "License Amendment Request for Technical Specifications Changes to Allow Use of Westinghouse 0.422-inch OD 14x14 VANTAGE+ Fuel," Docket 50-282 and 50-306, June 26, 2008, ADAMS Accession No. ML081820137.

2. Wadley, Michael D., Nuclear Management Company, letter to U.S. NRC, "Response to Acceptance Review Questions Re: License Amendment Request for Technical Specifications Changes to Allow Use of Westinghouse 0.422-inch OD 14x14 VANTAGE+ Fuel," Docket 50-282 and 50-306, August 4, 2008, ADAMS Accession No. ML082210260.

3. Wadley, Michael D., Northern States Power Company - Minnesota, letter to U.S. NRC, "Response to Request for Additional Information Regarding License Amendment Request for Technical Specifications Changes to Allow Use of Westinghouse 0.422-inch OD 14x14 VANTAGE+ Fuel," Docket 50-282 and 50-306, November 14, 2008, ADAMS Accession No. ML083190820.

4. Wadley, Michael D., Northern States Power Company - Minnesota, letter to U.S. NRC, "Response to Request for Additional Information Regarding License Amendment Request for Technical Specifications Changes to Allow Use of Westinghouse 0.422-inch OD 14x14 VANTAGE+ Fuel," Docket 50-282 and 50-306, January 30, 2009, ADAMS Accession No. ML090300684.

5. Wadley, Michael D., Northern States Power Company - Minnesota, letter to U.S. NRC, "Response to Request for Additional Information Regarding License Amendment Request for Technical Specifications Changes to Allow Use of Westinghouse 0.422-inch OD 14x14 VANTAGE+ Fuel," Docket 50-282 and 50-306, February 9, 2009, ADAMS Accession No. ML090410508.

6. Wadley, Michael D., Northern States Power Company - Minnesota, letter to U.S. NRC, "Response to Request for Additional Information Regarding License Amendment Request for Technical Specifications Changes to Allow Use of Westinghouse 0.422-inch OD 14x14 VANTAGE+ Fuel," Docket 50-282 and 50-306, February 20, 2009, ADAMS Accession No. ML090510691.

7. Cranston, Gregory, U.S. NRC, Memorandum to Lois James, U.S. NRC, "Audit Report for Nuclear Management Company – Prairie Island Nuclear Generating Plant Request for 422V+ Fuel Upgrade," Docket 50-282 and 50-306, October 24, 2008, ADAMS Accession No. ML082980156 (Non-publicly Available/Proprietary)

8. U.S. Nuclear Regulatory Commission, "Review Standard for Extended Power Uprates," RS-001, Revision 0, December 24, 2003, ADAMS Accession No. ML033640024.

9. Westinghouse Electric Company, "VANTAGE+ Fuel Assembly Reference Core Report," WCAP-12610-P-A, April 30, 1995, ADAMS Accession Nos. ML020430250 (Part 1) and ML020430146 (Part 2).

10. Westinghouse Electric Company, "Westinghouse Improved Performance Analysis and Design Model," WCAP-15063-P-A, July 21, 2000, ADAMS Accession No. ML003735390.

11. Peralta, J. D., U.S. NRC, letter to B. F. Maurer, Westinghouse Electric Company, "Approval for Increase in Licensing Burnup Limit to 62,000 MWD/MTU," May 25, 2006, ADAMS Accession No. ML061420458.

12. Westinghouse Electric Company, "Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel," WCAP-13589-A, March 31, 1995, ADAMS Accession No. ML080630348.

13. Westinghouse Electric Company, "Verification Testing and Analysis of the 17x17 Optimized Fuel Assembly," WCAP-9401-P-A, September 18, 1981, ADAMS Accession No. ML090280466.

14. Friedland, A. J. and Ray, S., Westinghouse Electric Corporation, "Revised Thermal Design Procedure," WCAP-11397-P-A, April 30, 1989, ADAMS Accession No. ML080630437.

15. Caldon, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the Leading Edge Flow Meter System," Engineering Report 80-P, March 1997, and U.S. NRC Approving Safety Evaluation Report, March 1999.

16. Caldon, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the Leading Edge Flow Meter Check [LEFM] or LEFM CheckPlus System," Engineering Report 157-P, October 2001, and U.S. NRC Approving Safety Evaluation, December 2001, ADAMS Accession No. ML013540256.

17. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report on the Applicability of WRB-1 to Westinghouse 14x14 and 15x15 OFA," June 29, 1984, ADAMS Accession No. ML083180570.

18. Sung, Y. X., Schueren, P., and Meliksetian, A., Westinghouse Electric Company, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," WCAP-15306-NP-A, October 31, 1999, ADAMS Accession No. ML993160096.

19. Westinghouse Electric Company, "Westinghouse Fuel Criteria Evaluation Process," WCAP-12488-A, October 1994, ADAMS Accession No. ML020430107.

20. Westinghouse Electric Company, "Fuel Criterion Evaluation Process Notification of Revision to 14x14 422 VANTAGE+ Design," October 27, 2005, ADAMS Accession No. ML053070066.

21. Not Used.

22. Westinghouse Electric Company, "TWINKLE – A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A, January 1975, ADAMS Accession No. ML080650324.

23. Westinghouse Electric Company, "FACTRAN – A Fortran IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989, ADAMS Accession No. ML080630436.

24. Westinghouse Electric Company, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," WCAP-14882-P-A, April 1999.

25. Chawla, Mahesh L., U.S. NRC, letter to Palmisano, T. J., Nuclear Management Company, "Prairie Island Nuclear Generating Plant, Units 1 and 2 Issuance of Amendments RE: Incorporate Large-Break Loss of Coolant Accident Analysis Using ASTRUM," Dockets 50-282 and 50-306, June 28, 2007, ADAMS Accession No. ML071230789.

26. Westinghouse Electric Corporation, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method," WCAP-16009-P-A, 2005.

27. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report for NOTRUMP, A Westinghouse Evaluation Model for Analyzing Small Break LOCAs," April 18, 1985, ADAMS Accession No. ML083520343.

28. Westinghouse Electric Company, "Core Uncovery Due to Loop Seal Re-Plugging During Post-LOCA Recovery," NSD-NRC-97-5092," 1997.

29. Westinghouse Electric Company, "Westinghouse ECCS Model Summary," WCAP-8339-NP-A, July 1974.

30. Westinghouse Electric Company, "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272, March 1978, ADAMS Accession No. ML051390150.

31. U.S. Nuclear Regulatory Commission, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," NUREG-1793, September 2004, ADAMS Accession No. ML043570339.

32. Collins, Daniel S., letter to Bischoff, G., Westinghouse Electric Company, "Suspension of NRC Approval for use of Westinghouse Topical Report CENPD-254-P, 'Post-LOCA Long-Term Cooling Model,' due to Discovery of Non-conservative Modeling Assumptions During Calculations Audit," Project Docket 0694, November 23, 2005, ADAMS Accession No. ML053220569.

33. Prairie Island Updated Safety Analysis Report, USAR, Rev. 29, May 2007.

34. Prairie Island Nuclear Generating Plant Units 1 and 2 – Response to Request for Additional Information regarding License Amendment Request for Technical Specifications Changes to Allow Use of Westinghouse 422V+ Fuel (TAC Nos. MD9142 and MD9143) – Letter to U. S. Nuclear Regulatory Commission, March 12, 2009 (ADAMS Accession No. ML090721088).

35. Prairie Island Nuclear Generating Plant Units 1 and 2 – Clarification of Response Regarding License Amendment Request for Technical Specifications Changes to Allow

Use of Westinghouse 0.422-inch OD 14x14 VANTAGE+ Fuel (TAC Nos. MD9142 and MD9143) – Letter to U. S. Nuclear Regulatory Commission, May 4, 2009 (ADAMS Accession No. ML091310384).

Principal Contributors:

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Date: July 1, 2009

July 1, 2009

Mr. Michael D. Wadley
Site Vice President
Prairie Island Nuclear Generating Plant
Northern States Power - Minnesota
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: TECHNICAL SPECIFICATIONS CHANGES TO ALLOW USE OF WESTINGHOUSE 0.422-INCH OD 14X14 VANTAGE+ FUEL (TAC NOS. MD9142 AND MD9143)

Dear Mr. Wadley:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 192 to Facility Operating License No. DPR-42 and Amendment No. 181 to Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Units 1 and 2 (PINGP), respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated June 26, 2008, as supplemented by letters dated August 4, August 26, and November 14, 2008, and January 30, February 9, February 20, March 12, and May 4 (2 letters), 2009.

The amendments revise the PINGP TSs for the use of Westinghouse 422 VANTAGE+ nuclear fuel and make changes to certain references identified in the Design Features section of the TSs.

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

Sincerely,
/RA/
Thomas J. Wengert, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306
Enclosures:
1. Amendment No. 192 to DPR-42
2. Amendment No. 181 to DPR-60
3. Safety Evaluation

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