April 1, 2004

Mr. David A. Christian
Sr. Vice President and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Blvd.
Glen Allen, Virginia 23060-6711

SUBJECT: NORTH ANNA POWER STATION, UNIT 2 - ISSUANCE OF AMENDMENT RE: USE OF FRAMATOME ANP ADVANCED MARK-BW FUEL (TAC NO. MB4715)

Dear Mr. Christian:

The Commission has issued the enclosed Amendment No. 216 to Renewed Facility Operating License No. NPF-7 for the North Anna Power Station, Unit 2. This amendment changes the Technical Specifications (TS) in response to your letter dated March 28, 2002, as supplemented by letters dated May 13, June 19, July 9, July 25, August 2, August 16, and November 15, 2002, May 6, May 9, May 27, June 11 (2 letters), July 18, August 20, August 26, September 4, September 5, September 22, September 26 (2 letters), November 10, December 8, and December 17, 2003, and January 6, January 22 (2 letters), February 12, February 13, and March 1, 2004. The November 15, 2002, submittal replaced the submittals dated July 9, July 25, and August 16, 2002.

This amendment revises TS Sections 2.1, 4.2.1, and 5.6.5. This change allows Virginia Electric and Power Company to replace the existing Westinghouse fuel with Framatome Advanced Nuclear Power Advanced Mark-BW Fuel at the North Anna Power Station, Unit 2.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice. The NRC staff is continuing to review the proposed license amendment for the North Anna Power Station, Unit 1 under TAC No. MB4714.

Sincerely,

/RA/
Stephen Monarque, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-339

Enclosures:
1. Amendment No.   to NPF-7
2. Safety Evaluation

cc w/encls: See next page

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Stephen Monarque, Project Manager, Section 1
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2. Safety Evaluation
cc w/encls: See next page

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VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-339

NORTH ANNA POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 216
Renewed License No. NPF-7

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

   A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated March 28, 2002, as supplemented by letters dated May 13, June 19, July 9, July 25, August 2, August 16, and November 15, 2002, May 6, May 9, May 27, June 11 (2 letters), July 18, August 20, August 26, September 4, September 5, September 22, September 26 (2 letters), November 10, December 8, and December 17, 2003, and January 6, January 22 (2 letters), February 12, February 13, and March 1, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-7 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 216, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to the initiation of core onload during Refueling Outage 16 (Spring 2004). As part of this implementation, the licensee shall describe, in the Updated Final Safety Analysis Report (UFSAR), that the thermo-mechanical design of Framatome ANP Advanced Mark-BW fuel design has been evaluated up to a peak rod burnup of 60,000 MWD/MTU for use at North Anna, Unit 2, as set forth in the NRC staff's safety evaluation dated April , 2004. These descriptions shall be reflected in the next update of the UFSAR submitted to the NRC pursuant to 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

John A. Nakoski, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance:

ATTACHMENT TO

LICENSE AMENDMENT NO. 216 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-7

DOCKET NO. 50-339

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove Pages	Insert Pages
2.0-1	2.0-1
4.0-1	4.0-1
5.6-3	5.6-3
5.6-4	5.6-4
5.6-5	5.6-5

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 216 TO RENEWED FACILITY OPERATING

LICENSE NO. NPF-7

VIRGINIA ELECTRIC AND POWER COMPANY

NORTH ANNA POWER STATION, UNIT 2

DOCKET NO. 50-339

1.0 INTRODUCTION

By letter dated March 28, 2002, as supplemented by letters dated May 13, June 19, July 9, July 25, August 2, August 16, and November 15, 2002, May 6, May 9, May 27, June 11 (2 letters), July 18, August 20, August 26, September 4, September 5, September 22, September 26 (2 letters), November 10, December 8, and December 17, 2003, and January 6, January 22 (2 letters), February 12, February 13, and March 1, 2004, Virginia Electric and Power Company (the licensee) requested changes to the Technical Specifications (TS) for North Anna Power Station, Unit 2. The November 15, 2002, submittal replaced the submittals dated July 9, July 25, and August 16, 2002. One of the supplements dated June 11, 2003, contained a new evaluation regarding the issue of no significant hazards consideration that replaced the evaluation contained in the original amendment request dated March 28, 2002, that had been noticed in the *Federal Register* on May 14, 2002 (67 FR 34496). Therefore, the amendment request was renoticed in the *Federal Register* on July 22, 2003 (68 FR 43397). The licensee's letters dated July 18, August 20, August 26, September 4, September 5, September 22, September 26 (2 letters), November 10, December 8, and December 17, 2003, and January 6, January 22 (2 letters), February 12, February 13, and March 1, 2004, provided clarifying information that did not change the scope of the proposed amendment as described in the revised notice of proposed action published in the *Federal Register* on July 22, 2003, and did not change the proposed no significant hazards consideration determination as published on July 22, 2003.

The requested changes would revise the North Anna TS to permit the use of Framatome Advanced Nuclear Power (ANP) Advanced Mark-BW fuel, including alloy M5, at North Anna Power Station, Unit 2. Specifically, the proposed amendment would revise TS Section 2.1.1, "Reactor Core SLs," to establish restrictions on the peak fuel centerline temperatures for the Westinghouse and Framatome fuel, TS Section 4.2.1, "Fuel Assemblies," to add alloy M5 to the list of cladding materials that may be present in the North Anna fuel assemblies, and TS Section 5.6.5, "Core Operating Limits Report," to include modifications of existing references and additional references that reflect the proposed changes.

2.0 REGULATORY EVALUATION

The NRC staff has identified the applicable regulatory requirements that the NRC staff considered in its review of the application. These requirements are listed below.

1. Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors,” specifies requirements for the acceptability of the emergency core cooling system (ECCS). 10 CFR 50.46(a)(1)(i) and 10 CFR 50.46(a)(1)(ii) specify alternative approaches to demonstrate compliance with the acceptance criteria of 10 CFR 50.46(b). Under the second approach, 10 CFR Part 50, Appendix K provides requirements for evaluating models for calculating ECCS performance. The acceptance criteria for calculated ECCS performance in the event of a postulated loss-of-coolant accident (LOCA) are as follows: 10 CFR 50.46 (b)(1), “Peak cladding temperature;” 10 CFR 50.46 (b)(2), “Maximum cladding oxidation;” 10 CFR 50.46 (b)(3), “Maximum hydrogen generation;” 10 CFR 50.46 (b)(4), “Coolable geometry;” and 10 CFR 50.46 (b)(5), “Long-term cooling.”

2. 10 CFR 50.55a, “Codes and standards,” requires that structures and components be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

3. 10 CFR Part 50, Appendix K, “ECCS Evaluation Models,” Section I.D.2, “Containment Pressure,” states “The containment pressure used for evaluating cooling effectiveness during reflood and spray cooling shall not exceed a pressure calculated conservatively for this purpose. The calculation shall include the effects of operation of all installed pressure-reducing systems and processes.”

4. General Design Criteria (GDC) 1, “Quality standards and records,” insofar as it requires that structures and components be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

5. GDC 2, “Design bases for protection against natural phenomena,” insofar as it requires that structures and components important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions.

6. GDC 4, “Environmental and dynamic effects design bases,” insofar as it requires that structures and components important to safety be designed to accommodate the effects of, and to be compatible with, the environmental conditions of normal and accident conditions.

7. GDC 10, “Reactor design,” insofar as it requires that the reactor core be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs).

8. GDC 14, “Reactor coolant pressure boundary,” insofar as it requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

9. GDC 15, “Reactor coolant system design,” insofar as it requires that the reactor coolant system (RCS) be designed with margin sufficient to ensure that the design conditions of the RCPB are not exceeded, during any condition of normal operation.

10. GDC 19, “Control room,” insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

11. GDC 25, “Protection system requirements for reactivity control malfunctions,” insofar as it requires that the protection system be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems.

12. GDC 27, “Combined reactivity control systems capability,” 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.

The NRC staff also considered the following guidance:

1. Regulatory Guide (RG) 1.157, “Best-Estimate Calculations of Emergency Core Cooling System Performance,” dated May 31, 1989, which provides guidance on methods acceptable to the NRC staff for realistic or best-estimate calculations of ECCS performance during a LOCA. In particular, Section 3.12.1, “Containment Pressure,” states, “The containment pressure used for evaluating cooling effectiveness during the post-blowdown phase of a loss-of-coolant accident should be calculated in a best-estimate manner and should include the effects of containment heat sinks. The calculation should include the effects of all pressure-reducing equipment assumed to be available. Best-estimate models will be considered to be acceptable provided their technical basis is demonstrated with appropriate data and analyses.”

2. NUREG-0800, “Standard Review Plan” (SRP), Section 6.2.1.5, “Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance,” Branch Technical CSB 6-1, “Minimum Containment Pressure Model for PWR [pressurized-water reactor] ECCS Performance Evaluation,” which provides guidance for complying with Appendix K to 10 CFR Part 50, Section I.D.2.

3. NUREG-0800, SRP Section 4.2, “Fuel System Design,” Revision 2, U.S. Nuclear Regulatory Commission, July 1981, which provides guidance for complying with 10 CFR Part 50, Appendix A, GDC 10.

4. USNRC Generic Letter (GL) 88-16, “Removal of Cycle-Specific Parameter Limits From Technical Specifications,” dated October 3, 1988. GL 88-16 provides guidance for relocating cycle-specific parameter limits from the TS to a Core Operating Limits Report (COLR). This guidance outlines a method for a licensee to implement a COLR to include cycle-specific parameter limits that are established using an NRC-approved methodology. Under GL 88-16, NRC-approved analytical methods used to determine

the COLR cycle-specific parameters are to be identified in the Administrative Controls section of the TS.

3.0 TECHNICAL EVALUATION

The NRC staff has reviewed the licensee's regulatory and technical analyses in support of its proposed license amendment, which are described in the licensee's submittal. Included as part of this evaluation are a variety of topics that were reviewed by the NRC staff. These topics are the containment pressure, the LOCA analyses, the structural integrity of the Framatome ANP Advanced Mark-BW fuel assembly and supporting reactor internals, the design-basis accidents, the spent fuel pool cooling, and the thermal-hydraulic analysis and mechanical design evaluation. The NRC staff's detailed evaluation is below.

3.1 Proposed TS Changes

The specific changes requested by the licensee are as follows:

1. TS Section 2.1.1.2 would be revised in its entirety to state, "The peak fuel centerline temperature shall be maintained < 5080^0 F, decreasing by 58^0 F per 10,000 MWD/MTU of burnup, for Westinghouse fuel and < 5173^0 F, decreasing by 65^0 F per 10,000 MWD/MTU of burnup, for Framatome fuel."

2. TS Section 4.2.1, "Fuel Assemblies," would be revised to allow for the use of M5 fuel rods.

3. TS Section 5.6.5, "Core Operating Limits Report (COLR)," would be revised to permit the use of NRC staff-approved LOCA methodologies and associated topical reports.

3.2 Minimum Containment Pressure

The containment pressure has an important effect on the LOCA calculations by influencing the rate of core reflooding. In this case, minimizing the containment pressure reduces the reflooding rate and therefore increases the peak cladding temperature (PCT). As such, the NRC staff reviewed the licensee's methodology for calculating containment pressure during a postulated LOCA at North Anna. The licensee presented these calculations in its submittal dated May 6, 2003. These pressure calculations were supplemented by supporting information in letters dated September 26, November 10, and December 8, 2003, and January 6, and January 22, 2004. The pressure calculations followed, in general, the methods described in Framatome ANP Topical Report EMF-2103 (P) Revision 0, "Realistic Large-Break Loss-of-Coolant Accident Methodology for Pressurized-Water Reactors," dated September 15, 2003. Differences with these methods are discussed in this section.

3.2.1 ICECON

The S-RELAP5 computer code, which performs the LOCA calculations, incorporates the containment computer code ICECON that was originally developed by Exxon Nuclear Company, Inc. ICECON is described in Siemens Power Corporation Topical Report EMF-CC-39(P), Revision 2, "ICECON: A Computer Program Used to Calculate Containment Back Pressure for LOCA Analysis (Including Ice Condenser Plants)," dated November 30, 1999. ICECON is based on the CONTEMPT/LT-022 computer code described in Aerojet

Nuclear Company Interim Report, I-214-74-12.1, “CONTEMPT-LT Users Manual,” dated August 1973. The NRC documented its approval of ICECON in an safety evaluation (SE) dated June 30, 1978. This SE stated that the NRC staff previously reviewed the Exxon ECCS evaluation model and published an SE dated September 11, 1975. The NRC staff’s SE dated September 11, 1975, concluded that Exxon’s containment pressure model for dry containments was acceptable for LOCA calculations. The June 30, 1978, SE approved the extension of this model to ice condenser containments as the ICECON code. The ICECON topical report states that ICECON can be used to model PWR dry, ice condenser, and subatmospheric containments. North Anna, Unit 2, has a subatmospheric containment.

Topical Report EMF-2102 (P) Revision 0, “S-RELAP5: Code Verification and Validation,” dated August 31, 2001, documented the assessment of ICECON against the GOTHIC 6.1 containment computer code in order to verify the capability of the S-RELAP5 Code, which incorporates ICECON, to calculate the LOCA phenomena in facilities using dry or ice condenser containments. GOTHIC 6.1 was developed by Numerical Applications Incorporated for the Electric Power Research Institute (EPRI) and has undergone extensive validation against data and analytic solutions. Topical Report EMF-2102 showed that ICECON compared well against the GOTHIC 6.1 containment computer code for a Westinghouse three-loop reactor with a dry containment.

Since the S-RELAP5 containment model is ICECON, which has previously been accepted by the NRC staff for LOCA minimum containment pressure calculations, and the licensee has demonstrated acceptable results in benchmarking ICECON to another well-validated containment computer code, GOTHIC 6.1, the NRC staff finds the use of ICECON by the licensee to be acceptable for realistic large-break (RLB) LOCA calculations.

CONTEMPT, as described above, is the basis for the ICECON code used by the licensee. An NRC staff study dated August 31, 2003, has shown that the CONTEMPT code containment spray model “tends to reduce pressure more rapidly than the data indicates.” Therefore, the containment spray mode is conservative for minimum pressure calculations and is acceptable for such calculations.

3.2.2 Calculating Containment Pressure

The licensee has chosen a combination of two approaches for calculating the containment pressure during a LOCA. This combination consists of both: (1) a statistical calculation and (2) the requirements of 10 CFR Part 50, Appendix K and the guidance of SRP Section 6.2.1.5, CSB 6-1. Using this combination, some of the input parameters are treated statistically while most are chosen in a manner that complies with the guidance of SRP Section 6.2.1.5, CSB 6-1 in order to underestimate the containment pressure. This methodology is acceptable since it complies with 10 CFR Part 50, Appendix K and results in an overall conservative calculation of containment pressure relative to the guidance of RG 1.157.

By letter dated November 10, 2003, the licensee described how the input for the containment model is selected. Two input parameters are sampled and treated statistically. These parameters are the containment volume and the initial temperature of the containment atmosphere. The containment volume is sampled over a range from nominal volume to the containment empty volume. The licensee described the containment empty volume as the sum of the volume of the containment dome and the volume of the containment cylinder, that is, the volume within the containment with no internal equipment or structures included. Since the pressure decreases with increasing containment volume, this range ensures that the selection of the containment volume is either best estimate (if the nominal containment volume is sampled) or conservative.

The initial containment air temperature is also treated statistically and is sampled over a reasonable range of values from 84.5° F to a maximum temperature of 121.5° F. The initial containment pressure is indirectly determined in the analyses by a procedure that uses this range of initial containment air temperature and North Anna TS Figure 3.6.4-1, “Containment Air Partial Pressure Versus Service Water Temperature.” The licensee stated that ranging containment temperature and volume encompasses normal variations in containment pressure. The NRC staff agrees that the licensee’s method does encompass normal variations and finds this to be acceptable.

In its submittal dated November 10, 2003, the licensee stated that it assumes a relative humidity of 100 percent. The licensee stated that the containment pressure is not sensitive to this parameter. The NRC staff agrees that the 100-percent value is a conservative number. The licensee treated the other parameters that affect containment LOCA pressure in accordance with SRP Section 6.2.1.5. This treatment assures a conservative bias to these parameters.

3.2.3 Containment Passive and Active Heat Sinks

The licensee’s analysis for the subatmospheric containment at North Anna has identified the heat sinks within containment that could influence containment pressure. This follows the guidance of SRP Section 6.2.1.5, CSB-6-1, which classifies two types of heat sinks as passive and active. The licensee’s analysis increases the nominal area of the passive or structural heat sinks by 3 percent and conservatively neglects the paint on all painted surfaces. Because the painted surfaces are a source of thermal resistance, this tends to overestimate the heat transfer to these passive heat sinks and results in minimizing the containment pressure.

The licensee identified the containment spray systems as the active heat sinks for North Anna, Unit 2. The North Anna subatmospheric design utilizes both a quench spray system and a recirculation spray system. The licensee did not apply the single active failure criterion to these systems. This is in agreement with the guidance of SRP Section 6.2.1.5, CSB 6-1, since this tends to minimize the pressure. The licensee stated that these active systems are assumed to operate at maximum efficiency in order to minimize the pressure.

An important consideration in determining the containment pressure is the heat transfer coefficient between the containment atmosphere and the passive heat sinks. The licensee uses the Uchida correlation with a multiplier of 1.7 for the blowdown and post-blowdown phases of the LOCA. The guidance of SRP Section 6.2.1.5, CSB 6-1 recommends the use of four times the value obtained from the Tagami condensation heat transfer correlation for the blowdown phase of the accident and the use of a multiplier of 1.2 times the value obtained from the Uchida correlation for the post-blowdown phase. The Uchida correlation models free convective condensation heat transfer. This specific model of condensation heat transfer produces a lower bound to the expected heat transfer since air currents caused by the sprays and general turbulence of the containment atmosphere will cause the heat transfer to be greater than that due only to natural circulation. The multipliers, such as the 1.7 value proposed by the licensee or the 1.2 value proposed in SRP Section 6.2.1.5, CSB 6-1, compensate for the use of the Uchida correlation when performing minimum pressure calculations to maximize the heat transfer for conservatism.

By letter dated December 8, 2003, the licensee provided a comparison of the results between the North Anna Updated Final Safety Analysis Report (UFSAR), 10 CFR Part 50, Appendix K minimum containment pressure calculation, and the North Anna ICECON containment pressure calculation. The licensee's comparison demonstrated that, using the same input assumptions, ICECON calculated a lower and thus more conservative containment pressure than the current UFSAR analysis. On this basis, the NRC staff finds the licensee's use of the Uchida correlation with a 1.7 multiplier for the blowdown and reflood portions of the accident to be acceptable. Independent NRC staff calculations have shown, however, that the Uchida correlation with a 1.7 multiplier does not necessarily give the most conservative result for other containment types. Therefore, the use of the Uchida correlation with a 1.7 multiplier for the blowdown and reflood portions of the accident is approved only for North Anna, Unit 2, RLBLOCA containment analyses.

The active North Anna heat sinks are the quench spray system and the recirculatuion spray system. The quench spray system takes suction from the refueling water storage tank (RWST), which is assumed to be at a temperature of 45° F. A lower RWST temperature is more conservative when determining the minimum containment pressure. North Anna TS Surveillance Requirement (SR) 3.5.4.1 limits the RWST temperature to greater than or equal to 40° F and less than or equal to 50° F. Since the licensee used a value of 45° F, rather than the more conservative lower bound temperature of 40° F, the NRC staff requested the licensee to either reanalyze the RLBLOCA at an RWST temperature of 40° F, revise the minimum TS value to 45° F, add a PCT penalty based on the difference between the values of the PCT determined for 45° F and 40° F, or provide a justification for using the 45° F temperature. By letter dated January 6, 2004, the licensee determined the difference in PCT between the value obtained assuming an RWST temperature of 45° F and assuming an RWST temperature of 40° F. The licensee's analysis showed an increase for the limiting case of 8° F. As a result, the licensee will include an additional 8° F on the reported 95/95 PCT result in the first 10 CFR 50.46 (a)(3)(ii) report of large-break (LB) LOCA PCT effects that will be submitted to the NRC following operation with Framatome ANP Advanced Mark-BW fuel. In addition, the licensee has stated that future North Anna RLBLOCA analyses will use either a biased lower bound or a sampled range that encompasses the allowable TS values. The licensee will incorporate this modeling change into the first RLBLOCA reanalysis performed following operation with Framatome ANP Advanced Mark-BW fuel. The NRC staff finds this approach acceptable, because the implementation of the 8° F PCT penalty adds conservatism to the PCT value and adequately compensates for the use of a RWST temperature greater than the TS lower bound.

3.2.4 Break Flow Analysis

An important aspect of the containment pressure calculation is the mass and energy release into the containment following an RCS pipe break. The same break flow assumed for the LOCA PCT calculation is used for the minimum containment pressure calculation. The break flow model is discussed in Section 4.3.3.2.7 of Framatome ANP Topical Report EMF-2103 (P). In addition, the homogeneous equilibrium critical flow model in S-RELAP5 is used. Since the licensee, in its submittal dated November 10, 2003, treated the break flow as a best-estimate parameter and uncertainty was appropriately taken into account, the NRC staff considers this to be acceptable.

Following the blowdown phase, ECCS water injected into the reactor vessel will be discharged from the break. The interaction of this water with the containment atmosphere can have an important effect on the containment pressure and the PCT. The licensee, in its letter dated November 10, 2003, stated that no ECCS injection is spilled directly to containment in the RLBLOCA model. Instead, as described in the licensee's letter dated January 22, 2004, the ECCS injection is mixed with the predominantly steam flow in the RCS. The licensee pointed out that there is no effective difference in the analysis between the mixing of the RCS steam and the ECCS injection flow occurring in the cold leg prior to discharge from the break and the mixing of these same fluids outside the cold leg. This is because the containment atmosphere is modeled as one volume. Therefore, the NRC staff finds the licensee's treatment of ECCS spillage to be acceptable.

3.2.5 Conclusion

Containment pressure has a significant effect on the PCT following a LOCA. Minimizing containment pressure is conservative for LOCA PCT calculations. The licensee's method of calculating containment pressure is a combination of a realistic approach considering uncertainty and a deterministic approach. Several input parameters are treated statistically (the containment volume, the initial temperature of the containment atmosphere, and the mass and energy discharged into the containment) while others are treated in a conservative manner relative to the guidance of RG 1.157 (containment heat sinks, both passive and active). Based on the NRC staff's review of the licensee's submittal and the conservative results obtained, as described above, the NRC staff finds the licensee's proposed method for calculating containment pressure for use in the RLBLOCA analysis acceptable.

3.3 LOCA Evaluation

3.3.1 LOCA Analysis

By letter dated May 6, 2003, as supplemented by letters dated May 27, August 20, September 4, September 5, and September 26, 2003, and February 12, 2004, the licensee provided LBLOCA and small-break (SB) LOCA analyses for the Framatome ANP Advanced Mark-BW fuel that will be implemented at North Anna, Unit 2. Because the change in inputs to consider a new fuel constitutes a significant change in the plant-specific LOCA methodologies, the licensee provided these LBLOCA and SBLOCA analyses results for the Westinghouse North Anna Improved Fuel (NAIF), the Framatome Advanced Mark-BW fuel, and the modeled Westinghouse surrogate fuel. The NRC staff then reviewed these analyses to ensure that the licensee met the requirements of 10 CFR 50.46(b). Because Framatome ANP does not have access to Westinghouse's proprietary fuel analyses, the use of a Westinghouse surrogate fuel model allows Framatome ANP to determine the impact of Westinghouse NAIF fuel at North Anna, Unit 2.

By letter dated May 6, 2003, as supplemented by letters dated August 20, September 5, and September 26, 2003, and February 12, 2004, the licensee provided the LBLOCA plant-specific analyses, using the methodology described in Topical Report EMF-2103 (P) Revision 0. Additionally, in its submittal dated May 21, 2003, the licensee submitted the results of the

10 CFR 50.46 Annual Report for North Anna, Unit 2, in order to address the bounding results for the NAIF fuel that will remain in the North Anna, Unit 2, core. The following table provides the licensee's LBLOCA analysis results.

TABLE 1 - LBLOCA

Limiting break Size/location	Double-ended guillotine/Pump Discharge (Cd=0.4)	Double-ended guillotine/Pump Discharge (2.59 ft^2)	Double-ended guillotine/Pump Discharge (2.59 ft^2)
Fuel Type	Westinghouse NAIF	Framatome Advanced Mark-BW	Modeled Westinghouse Surrogate Fuel
PCT	2152^o F	2096^o F	1995^o F
Maximum Local Oxidation	7.6%	4.0%	N/A
Maximum Total Core-wide Oxidation (All Fuel)	(<0.86%)	(<0.071%)	N/A

With regard to the SBLOCA analyses for North Anna, Unit 2, the licensee, in its submittal dated May 27, 2003, as supplemented by letter dated September 4, 2003, provided the plant-specific analyses using the methodology described in the NRC staff-approved Babcock and Wilcox Topical Report BAW-10168-P-A, Revision 3, "RSG LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," dated January 31, 1997. Additionally, in its letter dated May 21, 2003, the licensee provided the 10 CFR 50.46 Annual Report for North Anna, Unit 2. The licensee used the NRC staff-approved NOTRUMP SBLOCA evaluation model to determine the results of the SBLOCA analyses at North Anna, Unit 2. The following table provides the licensee's SBLOCA analysis results.

TABLE 2 - SBLOCA

Limiting Break Size/Location	3-inch Pump Discharge	3-inch Pump Discharge	3-inch Pump Discharge
Fuel Type	Westinghouse NAIF	Framatome Advanced Mark-BW	Modeled Westinghouse Surrogate Fuel
PCT	1689^o F	1618^o F	1404^o F
Maximum Local Oxidation	2.008%	0.16%	N/A
Maximum Total Core-wide Oxidation (All Fuel)	<1.00%	<0.02%	N/A

The licensee's LOCA analyses demonstrate that, consistent with the PCT results, the calculated post-LOCA oxidation for the NAIF fuel bounds the analyses for the Framatome ANP Advanced Mark-BW fuel.

At the NRC staff's request, the licensee also addressed the NRC staff's concern that the NAIF fuel may have a pre-existing oxidation condition that needed to be considered in estimating the maximum local oxidation that would occur in the event of a LOCA. In its supplemental letter dated August 20, 2003, the licensee provided its response to the NRC staff's concern by providing a reference to its cycle-specific fuel rod design criteria verification process. As such, the licensee has determined that the value of the calculated pre-transient oxidation for the NAIF fuel is sufficiently small so that the sum of the pre-transient and post-LOCA oxidation for the NAIF fuel is less than 17 percent. Based on the results of the analyses identified above, the NRC staff concludes that the licensee's LOCA analyses for North Anna, Unit 2, addresses the total LOCA oxidation and also meets the oxidation criterion of less than or equal to 17 percent of the total cladding thickness before oxidation as required by 10 CFR 50.46(b)(2).

Additionally, the NRC staff also notes that the pre-existing oxidation of the fuel is not expected to contribute to the LOCA maximum core-wide hydrogen generation. Therefore, the NRC staff concludes that the core-wide hydrogen generation analyses results reported above demonstrates that North Anna, Unit 2, meets the core-wide hydrogen generation criterion of 10 CFR 50.46 (b)(3).

As discussed above, the licensee has performed LBLOCA and SBLOCA analyses for North Anna, Unit 2, using LBLOCA and SBLOCA methodologies approved for North Anna, Unit 2. The licensee's LBLOCA and SBLOCA calculations demonstrated that the values for the PCT are 2152^{o} F and 2096^{o} F for NAIF and Framatome Advanced Mark-BW fuels, respectively, for the LBLOCA and 1689^{o} F for NAIF for the SBLOCA. Additionally, the licensee determined the maximum local oxidation values for the LBLOCA to be 7.6 percent plus pre-existing oxidation value and 4 percent, for NAIF and Framatome Advanced Mark-BW fuels, respectively, while the maximum local oxidation values for the SBLOCA were found to be 2.008 percent plus pre-existing oxidation value and 0.16 percent for NAIF and Framatome Advanced Mark-BW fuels, respectively. Finally, the licensee determined the core-wide hydrogen generation value for LBLOCA to be 0.86 percent and the SBLOCA value to be less than 1.0 percent for NAIF. These calculated values are less than the limits specified in 10 CFR 50.46(b) (1)-(3), which requires the PCT to be less than 2200^{o} F, the maximum cladding oxidation to be less than 17 percent, and the maximum hydrogen generation to be less than 1.0 percent. As a result, the licensee has demonstrated compliance with 10 CFR 50.46(b)(1)-(3). Additionally, the licensee, in Section 3.3.10, has demonstrated compliance with 10 CFR 50.46(b)(5). In as much as no other consideration affects North Anna core geometry, this assures that the North Anna, Unit 2, core will remain amenable to cooling as required by 10 CFR 50.46(b)(4).

In summary, the NRC staff concludes that the licensee's LOCA analyses were performed with LOCA methodologies that apply to North Anna, Unit 2, and demonstrate that North Anna, Unit 2, complies with the requirements of 10 CFR 50.46 (b)(1)-(5). Therefore, the NRC staff finds the licensee's LOCA analyses acceptable.

3.3.2 Mixed Core LOCA Analyses

In its letter dated May 6, 2003, the licensee provided LBLOCA analyses for the mixed core consisting of Framatome ANP Advanced Mark-BW fuel and Westinghouse NAIF fuel. Using the Framatome Topical Report EMF-2103 (P) model, the licensee proposed to analyze the Framatome ANP Advanced Mark-BW fuel with the North Anna, Unit 2, LOCA methodologies. Additionally, the licensee proposed to analyze the Westinghouse NAIF fuel with the present North Anna, Unit 2, methodologies using Westinghouse models. The licensee chose this approach because neither vendor's methodologies were qualified for application to the other vendor's fuel. However, during the review of this mixed-core methodology, the NRC staff became concerned about the geometrical differences between the Framatome ANP Advanced Mark-BW fuel and the NAIF fuel. The NRC staff's concern was based on the NRC's SE for Topical Report BAW-10227P-A, which concluded that because of the close similarity of M5 to Zircaloy, the effects of the differences between the cladding types on neighboring bundles would not be significant as long as the bundle geometries, including fuel and assembly dimensions and material surfaces, were alike. Due to these geometrical differences, the NRC staff requested that the licensee account for the fuel differences in the North Anna, Unit 2, LOCA analyses.

By letter dated May 6, 2003, as supplemented by letters dated August 20, 2003, and February 12, 2004, the licensee described a surrogate technique it had used to account for the mixed core containing both fuel types in the LBLOCA and SBLOCA analyses. For the mixed core, the licensee determined that the flow rate to Framatome ANP Advanced Mark-BW fuel would be reduced. This is accounted for in Framatome Topical Report EMF-2103 (P). In addition, as reflected in the results for the surrogate fuel, the licensee determined that the flow rate to the co-resident pre-burned Westinghouse NAIF fuel would increase. In its 10 CFR 50.46 Annual Report for North Anna, Unit 2, dated May 21, 2003, the licensee's analyses indicated that the results for the Westinghouse fuel bound both the Framatome ANP Advanced Mark-BW fuel and surrogate fuel results. Thus, the NRC staff finds that the licensee's proposed technique to account for the mixed core is acceptable because it directly accounts for the mixed core penalty in the results for Westinghouse NAIF fuel, and the licensee's technique reflects a benefit in the results for the surrogate fuel. Furthermore, the NRC staff finds that continuing to use the results as reported in the 10 CFR 50.46 Annual Report for the Westinghouse NAIF is acceptable because it accounts for the Westinghouse NAIF in a conservative fashion, by not including a mixed-core benefit. However, because the mixed core penalty for the Framatome ANP Advanced Mark-BW fuel and the PCT and oxidation values for the NAIF fuel were calculated using different methodologies, neither methodology correctly reflects the difference in results between the fuels. Therefore, pursuant to 10 CFR 50.46(a)(3)(ii), the licensee must track both LOCA methodologies and both fuel types in its 10 CFR 50.46 Annual Reports for the ECCS Evaluation Model Changes for North Anna, Unit 2.

3.3.3 Overall Applicability of LOCA Analysis Methodologies

In its letter dated September 5, 2003, the licensee provided a statement showing that it has ongoing processes with Framatome ANP for the purpose of assuring that the ranges and values of input parameters for the LOCA analysis, described in Topical Report EMF-2103 (P), bound the ranges and values of the as-operated plant values for those parameters. The licensee provided this information to show that it would properly model North Anna, Unit 2, the reported results would specifically represent the ECCS performance for North Anna, Unit 2, and these results would be within the applicability range of the model. Therefore, the NRC staff concludes that, in applying the LOCA methodology described in Topical Report EMF-2103 (P), the licensee will meet the requirements of 10 CFR 50.46(c).

By letter dated January 22, 2003, as supplemented by letter dated February 12, 2004, the licensee reported that certain rod bow effects had been inadvertently omitted in the licensee's fuel assessments, including the LOCA analyses. The licensee indicated that the effects associated with the Framatome ANP Advanced Mark-BW fuel assembly bow have been accommodated in the LBLOCA and SBLOCA analyses for both the resident NAIF fuel and the Framatome ANP Advanced Mark-BW fuel. The NRC staff concludes that the licensee's approach will assure that the rod bow effects issue will be bounded and is, therefore, acceptable.

3.3.4 Use of the Forslund Rohsenow Correlation

The NRC staff's SE dated April 9, 2003, for Topical Report EMF-2103 (P) questioned the validity of the use of the Forslund Rohsenow correlation. Subsequently, during its review of this amendment request, the NRC staff requested additional information regarding use of the Forslund Rohsenow correlation in the LBLOCA methodology. By letter dated September 26, 2003, the licensee provided the results of a sensitivity study quantifying the effect of the Forslund Rohsenow correlation on the result of North Anna, Unit 2, LBLOCA analyses. The results indicate that using the RLBLOCA methodology causes the Forslund Rohsenow correlation to have a significant (per 10 CFR 50.46) nonconservative effect on the North Anna, Unit 2 LBLOCA analyses. In a letter dated November 10, 2003, the licensee proposed to compensate for this nonconservatism by adding a 64^{o} F penalty to the PCT that was calculated with the methodology described in Topical Report EMF-2103 (P). Additionally, the licensee committed to documenting this PCT penalty in the North Anna UFSAR. The licensee generated this penalty by disabling the Forslund Rohsenow correlation to eliminate its nonconservative effect. The NRC staff finds that this provision is justified and acceptable. The analyses also exhibited a significant increase in time to quench. As such, the NRC staff requested that the licensee address the effect of the extended time to quench on oxidation results. The licensee addressed this concern in the November 10, 2003, letter by identifying the LBLOCA oxidation associated with the "non-Forslund Rohsenow" case in the sensitivity study as the North Anna, Unit 2, oxidation value of record. The NRC staff finds that the licensee's actions are acceptable.

3.3.5 Inclusion of Radiative Effects in Convective Heat Transfer Coefficient

The RLBLOCA methodology described in Topical Report EMF-2103 (P) does not provide radiative heat transfer correlations. Instead, the convective heat transfer correlations in the methodology are derived from empirical data that include a significant amount of radiative heat transfer. By including a significant amount of radiative heat transfer in the calculation of total heat transfer for a plant that does not have a significant amount of post-LOCA radiative heat transfer, a licensee could significantly overestimate the total post-LOCA radiative heat transfer for its plant. As a result, the licensee would nonconservatively underestimate the PCT.

To address the NRC staff's concern about this issue, the licensee, in its submittal dated January 22, 2003, provided calculations that compared the estimated radiative contribution to heat transfer for the North Anna, Unit 2, core to the radiative contribution to heat transfer for representative tests upon which the RLBLOCA methodology heat transfer models are based. The comparison indicates that the radiative heat transfer expected for this core is greater than the amount calculated using the RLBLOCA methodology for North Anna, Unit 2. The NRC staff concludes from this that the RLBLOCA methodology conservatively calculates less post-LOCA radiative heat transfer than would be expected for the core at North Anna, Unit 2, and, as a result, the NRC staff finds this to be acceptable.

3.3.6 Pellet Fragmentation and Relocation

The proposed LBLOCA methodology described in Topical Report EMF-2103 (P) did not provide for calculation of fuel pellet relocation. As such, the NRC staff postulated that ignoring these effects could lead the RLBLOCA methodology to underestimate the limiting PCT and oxidation values.

By letter dated September 5, 2003, the licensee provided sensitivity studies justifying the omission of fuel pellet relocation from the LBLOCA methodology. These sensitivity studies were performed for conditions that were less severe than test data but nevertheless remain bounding for the conditions at North Anna, Unit 2. The NRC staff concludes that this comparison demonstrates the applicability of the sensitivity studies to North Anna, Unit 2.

3.3.7 Lower Limit in Ranging of Break Size in RLBLOCA Calculations

In response to the NRC staff's concern that the minimum split break size in the ranging of large breaks was well below the size that delineates large breaks from small breaks, the licensee, in its letter dated September 22, 2003, stated that the smallest break size used in the distribution was at a size boundary between large and small breaks "with the total break area defined as the sum from both break junctions." The NRC staff notes that since the friction loss coefficient at the break varies inversely with break size, the sum of the flows from the two small-break junctions might likely be less than the break flow at the smallest large-break junction.

The licensee's analyses indicated that during blowdown the core region became completely voided, thus exhibiting a large-break characteristic. Afterwards, the liquid level in the core region began to increase after the initiation of accumulator injection. Based on these results, the licensee concluded that SBLOCA phenomena did not occur in any of the cases that were run using the RLBLOCA methodology described in Topical Report EMF-2103 (P). Based on the licensee's analyses, the NRC staff concludes that for the RLBLOCA methodology the ranging of break size to a smaller size than is usually considered large is acceptable because the smallest break size analyses exhibited LBLOCA phenomena. The NRC staff also noted that this demonstration for the smaller break sizes is unique to the present LBLOCA analyses at North Anna, Unit 2, and possibly attributable to low containment back pressure and the plant-specific vessel internals design.

3.3.8 Counter Current Flow Warning

In a letter dated December 20, 2002, Framatome ANP committed to implement a counter current flow limitation (CCFL) warning in the Topical Report EMF-2103 (P) S-RELAP5 computer code in order to alert an analyst to a CCFL violation in the downcomer region should one occur. By letter dated August 20, 2003, the licensee discussed that a CCFL warning had been implemented in the RLBLOCA methodology in order to address this matter. Based on its review of this information, the NRC staff concludes that this provision is responsive to the issue and, therefore, is acceptable.

3.3.9 Slot Breaks at the Top of the Pipe

The NRC staff's SE dated April 9, 2003, for Topical Report EMF-2103 (P) identified that Framatome ANP had not adequately addressed the NRC staff's concerns regarding slot breaks at the top and side of the pipe. The SE on the topical report called for an evaluation of the effects of the deep loop seal on slot breaks whenever an analysis has been performed for a plant with loop seals with bottom elevations that are below the top of the core. The SE stated that this evaluation may be based on relevant engineering experience, and the evaluation should be documented in either the RLBLOCA guideline or plant-specific guideline file. North Anna, Unit 2, features a loop seal design similar to that which was identified in the NRC staff's concern. In its letter dated March 1, 2004, the licensee provided information to show that the concern is essentially fuel-independent, and that analyses and procedures addressing this concern already exist for North Anna, Unit 2. In this response, the licensee referred to a letter dated August 2, 2001, from the NRC staff to Framatome ANP. In the August 2, 2001, letter, the NRC staff approved of Framatome ANP's SBLOCA analyses that concluded the conditions leading to extended core uncovery would not develop for a significantly long period of time. In addition, plant-specific procedures instruct operators to begin a timely depressurization of the primary system before the onset of extended core uncovery. In its March 1, 2004, letter, the licensee also stated that the Emergency Operating Procedures (EOPs) at North Anna, Unit 2, were based on approved Westinghouse Owners Group (WOG) EOP guidelines and direct timely operator actions that would avoid the conditions for extended core uncovery. In its August 20, 2002, letter, the licensee indicated that the operator procedures and actions would be effective in LBLOCA scenarios because extended core uncovery would take a significant amount of time to develop. The licensee has concluded that the existing provisions continue to apply to the upcoming cycle of operation, because the extended core uncovery issue of concern is fuel-independent.

Based on its review of the information provided by the licensee, and as set forth above, the NRC staff concludes that the licensee's analysis has successfully met the conditions of the NRC staff's SE dated April 9, 2003, for Topical Report EMF-2103 (P). The resolution of this issue applies to the current North Anna, Unit 2, licensing basis and does not resolve the generic issue of slot breaks at the top and side of the pipe in the RLBLOCA methodology.

3.3.10 RWST Temperature

The NRC staff inquired about the effect of the RWST temperature on the North Anna, Unit 2, LOCA calculations and boron precipitation analyses. TS SR 3.5.4.1 requires the RWST temperature be between 40^{o} F and 50^{o} F. In its response dated August 20, 2003, the licensee stated that the temperature of the injected water from the RWST is treated as nominal or is biased high in RLBLOCA analyses. The licensee's response indicates that this is conservative for PCT and initial quench. The NRC staff requested that the licensee extend the LBLOCA results plots to beyond the point of stable and sustained quench. The licensee provided its response on September 22, 2003. The licensee's response demonstrated a quench. However, the response did not resolve an NRC staff concern that the quench may not be sustained after the switchover of ECCS pump suction to the ECCS sump, which could result in the ECCS temperature being 100^{o} F hotter than the RWST temperature. Subsequently, in its letter dated November 10, 2003, the licensee provided information demonstrating that after ECCS switchover the temperature of the injected ECCS water would not be so high as to reduce the low-pressure injection (LPI) suction head below the point that the LPI pump performance would be significantly degraded.

The NRC staff requested that the licensee address the effect of the 40^{o} F to 50^{o} F RWST water temperature on the boron precipitation analysis. In its letter dated September 5, 2003, the licensee provided information reflecting the present boron dilution analysis at North Anna, Unit 2. Because the existing boron precipitation analysis is independent of the type of fuel used at North Anna, Unit 2, the NRC staff concludes that existing boron precipitation analysis would not be affected by the introduction of Framatome ANP Advanced Mark-BW fuel to the North Anna, Unit 2, core. Therefore, the NRC staff concludes that the licensee has demonstrated compliance with 10 CFR 50.46 (b)(5). Additionally, the NRC staff has determined that the current approved boron precipitation analysis continues to apply to the use of Framatome ANP Advanced Mark-BW fuel.

3.3.11 Downcomer Boiling

By letter dated September 22, 2003, the licensee provided the results of an analysis it had performed that indicated downcomer boiling would not occur to the extent that it would significantly degrade core cooling in the first 640 seconds of an LBLOCA transient. The licensee's results indicate that after 640 seconds the reactor vessel inventory would continue to remain reasonably constant, thus exhibiting a rising trend. This is consistent with the downcomer boiling analysis results reported generically for the RLBLOCA analysis methodology described in Topical Report EMF-2103 (P). Therefore, the NRC staff finds this acceptable. The NRC staff is presently pursuing concerns related to downcomer boiling in a generic matter. If that review raises any concerns applicable to the LOCA analyses at North Anna, Unit 2, then the NRC staff will request the licensee to address these issues consistent with any generic resolution.

3.3.12 Vessel Internals

The NRC staff requested that the licensee provide descriptive information regarding the North Anna, Unit 2, vessel internals so that the NRC staff could verify that reactor vessel internals were properly represented in the LBLOCA and SBLOCA methodologies. By letter dated August 20, 2003, the licensee indicated that the reactor vessel internals were accounted for in the RLBLOCA methodology described in Topical Report EMF-2103 (P) and the SBLOCA methodology described in Topical Report BAW-10168-P-A. Therefore, the NRC staff finds that the licensee is properly representing the vessel internals in the North Anna, Unit 2, LBLOCA and SBLOCA methodologies.

3.4 <u>Structural Integrity of the Fuel Assembly and Supporting Reactor Internals</u>

By letter dated March 28, 2002, as supplemented by letters dated May 9, and June 11, 2003, the licensee evaluated the structural integrity of the Framatome ANP Advanced Mark-BW fuel assembly under normal operating, upset, and faulted conditions. Specifically, the licensee evaluated the structural adequacy of the holddown spring, guide thimbles, spacer grids, and top and bottom nozzles using the methodology described in the NRC staff-approved Topical Report BAW-10133P, "Mark-C Fuel Assembly LOCA-Seismic Analyses," dated October 30, 2000. In addition, the licensee calculated the stress limits for these components and found these limits to be below the criteria specified in the 1989 Edition of the American Society of

Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Subsection NG. The licensee indicated that this was consistent with the criteria outlined in the North Anna UFSAR. The licensee's submittal dated May 9, 2003, had identified the 1989 Edition of the ASME Code, Section III, Subsection NG as the Code of record that was used in the evaluation of the Framatome ANP Advanced Mark-BW fuel assembly.

The licensee stated in its March 28, 2002, application that it had performed a structural analysis of the Framatome ANP Advanced Mark-BW fuel, including an evaluation of guide thimble buckling. A detailed finite element analyses of the top and bottom nozzles had also been performed using the finite element program ANSYS. These analyses were determined to be within the allowable limits of the ASME Code. In its submittal dated May 9, 2003, the licensee provided a comparison between the maximum stress and the design-basis allowable stress for each critical component. Additionally, in this same submittal, the licensee provided a comparison between the fatigue usage factor and the design-basis allowable stress for the holdown spring, clamp bolt, and fuel rods. These comparisons are shown on Tables 3.1 and 3.2, respectively, of the licensee's submittal dated March 28, 2002. As shown on Table 3.2, the licensee demonstrated that the maximum cumulative usage factor (CUF) was less than 1.0 for the holddown spring for the design fuel service life under the loading cycles described in Table 3.3-1 of the March 28, 2002, submittal. Based on this information, the NRC staff finds the licensee's calculated maximum stresses and CUF to be below the allowable limits of ASME Code, Section III, Subsection NG and, therefore, acceptable.

The comparison of design data provided in Table 1.0-1 of the licensee's submittal dated March 28, 2002, indicates that the Framatome Advanced Mark-BW fuel is similar in design to the Westinghouse NAIF currently licensed and operated at North Anna, Unit 2. By letter dated May 9, 2003, the licensee provided test results in Table 1.1 that demonstrate that these two fuel assemblies are dynamically similar in stiffness, frequencies, and dampings. Furthermore, in its submittal dated June 11, 2003, the licensee indicated that the upper and lower core plate motions used in the fuel faulted analysis are not affected by the fuel assembly changes at North Anna. Based on this information, the NRC staff concludes that the use of Framatome Advanced Mark-BW fuel will not affect the structural responses to static or dynamic loads that were considered in the original analyses, and use of this fuel type will not have an adverse effect on its supporting internal components.

Based on its review of the information provided by the licensee, the NRC staff concludes that the maximum stress and fatigue usage factors at critical locations of the Framatome ANP Advanced Mark-BW fuel components are within the allowable design limits of ASME Code, Section III, Subsection NG. Furthermore, the NRC staff has determined that the use of the Framatome ANP Advanced Mark-BW fuel at North Anna, Unit 2, is acceptable with respect to the structural integrity of the fuel assembly and the supporting reactor internals.

3.5 Design-Basis Accidents

In support of the proposed changes, the licensee considered the impact of the license amendment upon the previously analyzed design-basis accidents having radiological consequences. In its submittal dated March 28, 2002, the licensee stated that the key design inputs that would impact the existing radiological analyses would either remain unchanged or the changes themselves would have no impact and, as such, the existing analyses would remain applicable for the use of Framatome ANP Advanced Mark-BW fuel at North Anna, Unit 2.

In its submittal, the licensee considered the LOCA, the fuel handling accident (FHA), the locked rotor accident (LRA), the main steamline break (MSLB), and the steam generator tube rupture (SGTR) in order to determine whether the accident analyses bounded the expected consequences regarding the use of Framatome ANP Advanced Mark-BW fuel. As part of these analyses, the licensee assessed key design inputs of the radiological analyses in order to identify those inputs that may potentially be impacted. These key design inputs and their changes are tabulated below.

Design Input	Change
Fuel theoretical density	Increasing from 95% to 96%
Peak rod average burnup	Unchanged at 60,000 MWD/MTU
Peak enthalpy rise hot channel factor	Unchanged at 1.65
Core average power	Unchanged at 102% of 2893 MWt
Enrichment	Unchanged at 4.6 w/o U-235
Event-specific cladding failure	Unchanged
Rod internal pressure (100 hours after S/D)	Unchanged at $\leq$1200 psig

The licensee stated that the increase in fuel theoretical density will not cause an increase in the radiological source terms assumed in the LOCA, FHA, and LRA analyses since the core inventory is a function of the core average thermal power, peak rod average burnup, and the peak enthalpy rise hot channel factor, and these items would remain unchanged.

The NRC staff concurs with the licensee's evaluation discussed in the previous paragraph. The NRC staff notes that the source terms used in analyzing the MSLB and SGTR events are based on the TS Section 3.4.16, "RCS specific activity" limiting condition for operation. As such, the previous analyses of the radiological consequences of these events are not affected by the proposed fuel change. The NRC staff also notes that the proposed change in fuel design will not impact any plant parameters that factor into the analysis of the transport of fission products from the core to the environment.

The NRC staff has reviewed the licensee's amendment request and, for the reasons set forth above, has determined that the proposed use of Framatome ANP Advanced Mark-BW fuel at North Anna, Unit 2, will not impact the previously analyzed radiological consequences of design-basis accidents. Since the design-basis accident analysis results were previously found to meet the acceptance criteria of 10 CFR Part 100 and 10 CFR Part 50, Appendix A, GDC-19, the proposed fuel change is acceptable with regard to the design-basis accident radiological consequences.

3.6 Thermal-Hydraulic Analysis and Mechanical Design Evaluation

The Framatome ANP Advanced Mark-BW fuel design is an evolution of the Framatome ANP Mark-BW fuel design and incorporates new design features including the TRAPPER bottom nozzle, Mid-Span Mixing Grids (MSMGs), a floating intermediate grid design, a quick disconnect top nozzle, and use of M5 material for the cladding, structural tubing, and grids. The NRC staff is concurrently reviewing the Framatome ANP Advanced Mark-BW design topical report for generic approval; however, since the fuel design is not currently approved for generic use, the North Anna review encompasses both the fuel design as well as plant-specific use of the fuel. The specific plant use of the fuel design includes analyzing the mixed-core parameters and the UFSAR non-LOCA transients.

The objectives of this fuel system safety review, as described in Section 4.2 of the SRP, are to provide assurance that: 1) the fuel system is not damaged as a result of normal operation and 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. A not damaged fuel system is defined as fuel rods that do not fail, fuel system dimensions that remain within operational tolerances, and functional capabilities that are not reduced below those assumed in the safety analyses. Fuel rod failure means that the fuel rod leaks, and that the first fission product barrier (the cladding) has been breached. Coolability, which is sometimes termed coolable geometry, means that the fuel assembly retains its rod-bundle geometrical configuration with adequate coolant channels to permit removal of residual heat even after an accident.

Implementation of the Framatome Advanced Mark-BW fuel design in the North Anna reactor core necessitates modifying three technical specifications. These modifications include changing the peak fuel centerline equation safety limit, changing the fuel assembly design description, and incorporating approved methods into the COLR references section.

3.6.1 Fuel Assembly Design

The Advanced Mark-BW fuel design is intended for use in Westinghouse three- and four-loop reactors, which use a 17 x 17 fuel rod array. The design is based on the Framatome ANP Mark-BW design and incorporates some additional features that have been proved through reactor use such as: the TRAPPER bottom nozzle, MSMGs, a floating intermediate grid design, and a low-pressure drop-quick disconnect top nozzle. The design also uses the M5 advanced alloy, which has been previously approved for cladding, structural tubing, and grids in Framatome Topical Report BAW-10227-P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," dated February 29, 2000. A thorough description of the quick connect/disconnect top nozzle, guide thimble and instrumentation tubing, spacer grids, MSMGs, debris filter bottom nozzle, and the fuel rods, along with their associated drawings, is provided in Section 1.0 of the licensee’s submittal dated March 28, 2002. In addition, the component test matrix is provided in Section 2.0 of Attachment 2 in this same submittal. Therefore, a satisfactory description of the fuel assembly has been provided for this review as per the guidance of SRP 4.2.

3.6.2 Advanced Material (M5) and Post Irradiation Examination (PIE)

The M5 material is an advanced zirconium alloy material that was reviewed and approved for use in Topical Report BAW-10227-P-A. M5 has many improved characteristics including: corrosion, hydrogen pickup, axial growth, and diametral creep.

The Advanced Mark-BW fuel design is an evolution of the Mark-BW fuel design. The in-reactor performance of the Mark-BW fuel design shows that it has positive performance. This performance is demonstrated through the use of a large number of fuel rods. The Advanced Mark-BW design and components were also subject to a range of tests included in section 2.0 of Attachment 2 of the licensee’s submittal dated March 28, 2002. This array of tests shows that the fuel design and components have been adequately tested.

By letter dated September 23, 2003, the NRC staff issued an exemption from the requirements of 10 CFR 50.44, 10 CFR 50.46, and Appendix K (68 FR 56338, dated September 30, 2003).

This exemption allowed the licensee to demonstrate the adequacy of its calculated ECCS performance given the use of advanced zirconium-based alloy M5 as the fuel cladding material at North Anna, Unit 2. The NRC staff's evaluation in the exemption is hereby incorporated by reference into this SE, pursuant to 10 CFR 50.32.

3.6.3 Mechanical Design Evaluation

The fuel system design bases should reflect the following four objectives as per SRP section 4.2: 1) the fuel system is not damaged as a result of normal operation and 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. In addition, the Advanced Mark-BW fuel should be compatible with the resident Westinghouse NAIF fuel assemblies when used in a mixed core as described in section 3.6.3.2 of this SE.

3.6.3.1 Lead Test Assembly (LTA) Experience

The LTA program, for confirming the irradiation behavior of the Advanced Mark-BW fuel assembly design, used four LTAs in locations where the LTAs saw near-peak core power conditions. The LTAs were irradiated for three cycles in the North Anna, Unit 1 reactor. During their core residency, two cycles were in high duty locations and, during the third cycle, the LTAs were placed on the core periphery. The core periphery is a hydraulic environment that has, in the past, resulted in flow-induced vibration failures in fuel assemblies at several reactors. The post irradiation examinations (PIEs) were performed after every irradiation cycle to confirm that the LTAs were operating as predicted. The PIEs performed were appropriate for confirming the performance of the fuel design and the results confirmed predictions; therefore, the LTA performance is acceptable.

3.6.3.2 Fuel Assembly Compatibility

Compatibility evaluations are performed to demonstrate that the dimensions and configurations of the new fuel assembly components are compatible at the interfaces between the fuel assembly and the other reactor components and fuel assemblies, and that functionality of all components is maintained. The Advanced Mark-BW fuel design was compared to the resident fuel to evaluate the compatibility of the mechanical interfaces between the two designs and between the new fuel and the reactor. The Advanced Mark-BW design was also evaluated for compatibility with the fuel shipping and handling apparatus. The NRC staff reviewed the compatibility evaluations as part of an audit. It was noted that tolerances were consistently combined and treated in a statistically appropriate manner. The evaluations demonstrated that the Advanced Mark-BW fuel assemblies will be compatible with the North Anna reactor and the Westinghouse NAIF resident fuel assemblies.

3.6.3.3 Structural Integrity

The analyses for the fuel holddown springs demonstrated that during all conditions considered, except for the pump overspeed transient, the fuel assembly remained in contact with the lower support plate. During the pump overspeed transient, the lift is small, and the force applied to the fuel assembly does not cause the holddown spring to deflect to the solid state. The holddown spring is not compressed to a solid height for any operating condition. Therefore, the fuel assembly holddown springs will perform their intended function during normal operating conditions.

The spacer grids are designed so that crushing does not occur under normal operation and operational base earthquake conditions. Additionally, under Safe Shutdown Earthquake (SSE) conditions the spacer grids function to maintain core internal geometry to ensure control rod insertion, and they ensure safe shutdown of the reactor by maintaining the overall structural integrity of the fuel assemblies and a coolable geometry within deformation limits for SSE and LOCA accidents. Refer to Framatome Topical Report BAW-10133-P-A, Revision 1, Addendum 1, "Mark-C Fuel Assembly LOCA-Seismic Analyses," dated October 30, 2000, and Babcock and Wilcox Topical Report BAW-10133-P-A, Revision 1, "Mark-C Fuel Assembly LOCA-Seismic Analyses," dated June 16, 1986. These evaluations demonstrate that the spacer grids can meet all of these design criteria.

Analysis of the top and bottom nozzles at end-of-life (EOL) shutdown conditions showed that when a scram load is applied, the top and bottom nozzles remain engaged with the reactor internals. The NRC staff's audit verified that under all operating conditions the top and bottom nozzles remain engaged with the reactor internals.

The guide thimbles for the resident fuel assemblies are smaller than the guide thimbles for the Advanced Mark-BW fuel design. Analysis has shown that the guide thimbles of the Advanced Mark-BW fuel design will not buckle under normal or transient conditions when control rod insertion is needed. Therefore, since the guide thimbles will remain intact and the guide thimbles are larger than the resident fuel, the TS SR 3.1.4.3 control rod drop time of 2.7 seconds will remain applicable for the Advanced Mark-BW fuel design.

Earthquakes and postulated pipe breaks in the RCS would result in external forces on the fuel assembly. During these events, fuel system coolability should be maintained and damage should not be so severe as to prevent control rod insertion when required. The design criteria for fuel assembly structural damage from external forces are divided into two categories:

> SSE - Ensure safe shutdown of the reactor by maintaining the overall structural integrity of the fuel assemblies, control rod insertibility, and a coolable geometry within the deformation limits consistent with the ECCS and safety analysis.
>
> LOCA or SSE and LOCA - Ensure safe shutdown of the reactor by maintaining the overall structural integrity of the fuel assemblies and a coolable geometry within deformation limits consistent with the ECCS and safety analysis.

The licensee used the methodology in Topical Reports BAW-10133-P-A, Revision 1, Addendum 1, and BAW-10133-P-A to perform generic evaluations of the structural damage from external forces. These analyses considered the horizontal and vertical impacts on the fuel assembly. These analyses included evaluations of the impact on the Advanced Mark-BW fuel design when it is located in a mixed core with the Westinghouse NAIF fuel under the worst core configurations. Various core-loading patterns and locations in the core were utilized for the mixed-core analysis impact. The results showed that the combined loads on the Advanced Mark-BW fuel assembly were small enough that coolable geometry is always maintained. The analyses demonstrated that coolable geometry can be maintained under all the analyzed conditions; therefore, it was demonstrated that the design criteria are met.

3.6.3.4 Fuel Rod Design

The fuel system design bases should reflect the following four objectives: 1) the fuel system is not damaged as a result of normal operation and 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.

3.6.3.4.1 Fuel Rod Cladding Stress

The stress analyses for the Framatome Advanced Mark-BW fuel design were reviewed as part of an NRC staff audit. Framatome Topical Report BAW-10239, "Advanced Mark-BW Fuel Assembly Mechanical Design Topical Report," defines the Framatome Advanced Mark-BW fuel design criterion for stress as "Stress intensities for the Advanced Mark-BW fuel assembly components shall be less than the stress limits based on the ASME Code, Section III criteria." This criteria conforms to the guidance of SRP 4.2. For the stress analyses calculations, the worst-case values in combination with the most limiting cycle conditions and most limiting transients were used to generate the most conservative results for each calculation. This deterministic method to obtain the most limiting stress value provides the most conservative stress value for each fuel assembly component. Positive margin to the design criteria was shown for each of the fuel assembly components; therefore, the design stresses are acceptable.

3.6.3.4.2 Fuel Rod Cladding Strain

The Advanced Mark-BW fuel rod transient strain (elastic plus plastic) limit is 1 percent for Conditions I and II events. This criterion is intended to preclude excessive cladding deformation during normal operation and AOOs. The analysis of the cladding strain uses the approved TACO3 Code, referenced in Babcock and Wilcox Topical Report BAW-10162-P-A, "TACO3 Fuel Pin Analysis Computer Code," dated December 13, 1989, to determine the cladding strain by evaluating the cladding circumferential changes before and after a linear heat rate (LHR) transient. The 1-percent strain limit corresponds to a transient LHR that is greater than the maximum transient the fuel rod is expected to experience for Condition I and II events. Therefore, the cladding strain is acceptable.

3.6.3.4.3 Fuel Rod Fatigue Usage

The Advanced Mark-BW maximum fuel rod fatigue usage factor is 0.9. The methodology used for determining the cladding fatigue is outlined in Topical Report BAW-10227-P-A. This methodology used a fuel rod life of 8 years and a vessel life of 20 years; therefore, the fuel rod will experience 20 percent of the number of transients that the vessel will. The analysis used all the Condition I and II events and one Condition III event to determine the total cladding fatigue usage factor. The maximum fatigue usage factor was determined to be well below the design criteria limit. Since the methodology is consistent with SRP 4.2 guidance, and the maximum fatigue is well below the design criteria limit, the licensee has demonstrated that the cladding fatigue acceptance criteria have been met.

3.6.3.4.4 Fuel Rod Cladding Creep Collapse

Framatome Topical Report BAW-10239 states that “The acceptance criterion is that the predicted creep collapse life of the fuel rod must exceed the maximum expected in-core life.” The SRP states that if axial gaps in the fuel pellet column occur due to densification, the cladding has the potential of collapsing into a gap. Because of the large local strains that accompany this process, collapsed cladding is assumed to fail.

The licensee used the approved creep collapse methodology referenced in Babcock and Wilcox Topical Report BAW-10084-P-A, Revision 3, "Program to Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse," dated July 18, 1995, to determine the potential for creep collapse of the Advanced Mark-BW fuel assembly design. This methodology uses conservative values to determine the creep collapse life of the fuel rod. Based on the approved definitions of creep collapse, the creep collapse lifetime was shown to be greater than 62 GWD/MT. Therefore, the Advanced Mark-BW fuel design is adequately designed to prevent creep collapse when used in the North Anna reactor core for a service life up to 62 GWD/MT.

3.6.3.4.5 Fuel Rod Cladding Corrosion

Framatome Topical Report BAW-10239 states that “The fuel rod cladding best-estimate corrosion shall not exceed 100 microns.” The SRP does not specify specific limits on cladding oxidation, hydrogen pickup, and crud but does specify that their effects should be accounted for in the thermal and mechanical analyses performed for the fuel. Corrosion for the M5 cladding material is accounted for based on a database established for the M5 cladding material from in-reactor performance. This is acceptable since it uses realistic data that are representative of the material and burnup limits of the Advanced Mark-BW fuel design.

Based on the data for M5 cladding material under prototypic irradiation conditions, the corrosion rates are well below the criterion. Since crud is included as part of the corrosion measurement, the crud is also limited and well within the total acceptable range. Hydrogen pickup is a function of oxidation. For example, a lower rate of oxidation results in a lower rate of hydrogen pickup. As such, Framatome ANP’s experimental data has demonstrated that the M5 cladding material will be below Framatome ANP’s manufacturing limit of 710 ppm. Therefore, the licensee has demonstrated that the corrosion, hydriding, and crud buildup for the Advanced Mark-BW fuel design has met the acceptance criteria.

3.6.3.4.6 Fuel Rod Shipping and Handling Loads

As described in the criterion of Framatome Topical Report BAW-10239, the fuel rod should withstand axial loading during shipment and handling without gaps forming between pellets in the fuel stack. The formation of gaps causes deformation of cladding material. Stainless steel springs in the upper plenum provide a pre-load on the fuel stack to prevent the formation of fuel pellet gaps. This design has been tested through extensive use and shown to be capable of maintaining the fuel stack without the formation of gaps.

3.6.3.4.7 Fuel Rod Fretting Wear

Framatome Topical Report BAW-10239 states that “The fuel assembly design shall be shown to provide sufficient support to limit fuel rod vibration and clad fretting wear.” This is because fuel rod vibration and clad fretting wear could cause fuel failures. The Advanced Mark-BW fuel design is an evolution of the Mark-BW fuel design. The in-reactor performance of the Mark-BW fuel design shows that it has positive performance in the fretting arena. This performance is demonstrated through the use of a large number of fuel rods. Similarly, the in-reactor performance of the Advanced Mark-BW LTAs produced positive results even when the LTAs were subjected to the hostile hydraulic environment of the core periphery. Out-of-core testing was also performed on the Advanced Mark-BW fuel design, including a 1000-hour endurance test. The results of the endurance test demonstrated that fuel rod wear was comparable to other currently approved fuel designs. These tests and data demonstrate that the Advanced Mark-BW fuel design has design features to assure that the design criteria for fretting will be met.

3.6.3.4.8 Fuel Rod Growth

Framatome Topical Report BAW-10239 states that “The fuel assembly-to-reactor internals gap allowance shall be designed to provide positive clearance during the assembly lifetime,” and “The fuel assembly top nozzle-to-fuel rod gap allowance shall be designed to provide positive clearance during the assembly lifetime.” The axial growth calculations were reviewed as part of the NRC staff audit of the Advanced Mark-BW fuel design. The audit showed that the evaluation of the Advanced Mark-BW fuel design used approved M5 growth models and the worst-case scenarios for calculating the clearances. The tolerances were combined in an appropriate manner and treated consistently. The lowest clearance values were obtained at EOL and, in all evaluations, positive clearance remained at EOL under the worst conditions. Therefore, the axial growth for the Advanced Mark-BW fuel design is acceptable.

3.6.3.4.9 Fuel Rod Internal Pressure

To minimize fuel rod failures, Framatome Topical Report BAW-10239 states that “the fuel system will not be damaged due to excessive internal pressure. Fuel rod internal pressure is limited to that which would cause the diametral gap to increase due to outward creep during steady-state operation and extensive departure from nucleate boiling (DNB) propagation to occur.” The analysis used the TACO3 code with the methodology approved in Babcock and Wilcox Topical Report BAW-10183-P-A, "Fuel Rod Gas Pressure Criterion (FRGPC)," dated July 24, 1995. The analysis used the bounding pin power envelope and axial flux shapes and included the use of the most limiting manufacturing variations. This bounding analysis demonstrated that the fuel rod internal pressure criteria were met.

3.6.3.4.10 Linear Heat Rate to Melt

To provide reasonable assurance that fuel melting is prevented, Framatome Topical Report BAW-10239 states “For a 95-percent probability at a 95-percent confidence level, fuel pellet centerline melting shall not occur for normal operation and AOOs.” SRP 4.2 states that this analysis should be performed for the maximum linear heat generation rate anywhere in the core, including all hot spots and hot channel factors, and should account for the effects of burnup and composition on the melting point. Framatome used the TACO3 computer code to determine the peak linear power that prevents centerline melt. This value was determined to be 21.9 kW/ft. This value is greater than the 18.0 kW/ft value predicted for the North Anna reactor core, as indicated in the   Chapter 4 USFAR analyses. This analysis was reviewed as part of the NRC staff’s audit and the results were confirmed.

3.6.4 Thermal-hydraulic Evaluation

The primary purpose of thermal-hydraulic analysis is to demonstrate acceptable thermal performance to ensure that fuel and clad integrity are maintained during normal operation and transients of moderate frequency. To achieve this purpose, design criteria have been established. The licensee's submittal dated March 28, 2002 states the following:

1) "During Condition I and II events, there must be at least a 95-percent probability with a 95-percent confidence level that the hot pin will not experience a DNB; and a 99.9-percent probability that DNB will not occur core-wide."

2) "During Condition I and II events, there must be at least a 95-percent probability with a 95-percent confidence level that no fuel rod will experience centerline melting."

The secondary purpose of the thermal-hydraulic analysis is to ensure hydraulic compatibility between the Advanced Mark-BW fuel and the Westinghouse NAIF fuel that is resident in the North Anna core.

3.6.4.1 Design Comparison

The resident fuel is the Westinghouse NAIF design. It is based on the Westinghouse 17x17 VANTAGE-5H design with debris-resistant features and ZIRLO cladding. The primary differences between the Advanced Mark-BW design and the resident fuel include the use of the M5 material, a non-mixing vane grid at the lower intermediate grid location, and three MSMGs.

3.6.4.2 Thermal-hydraulic Core Models and Inputs

The thermal-hydraulic analyses supporting the use of the Advanced Mark-BW fuel design used Framatome's approved LYNXT code described in Babcock and Wilcox Topical Report BAW-10156-P-A, Revision 1, "LYNXT: Core Transient Thermal-Hydraulic Program," dated August 31, 1993. The analyses contained in this section incorporated the effects of a measurement uncertainty recapture uprate of 1.7 percent from the current licensed power level. Although the analyses have been performed in support of the uprate, this licensing action does not request approval of the uprate; thus, this SE does not approve the thermal-hydraulic core models and inputs for the Advanced Mark-BW fuel design at the uprated conditions, nor does it imply any agreement with the results using the higher power level.

3.6.4.2.1 LYNXT Modeling

The LYNXT thermal-hydraulic analysis code is a single-pass code that employs crossflow methods to evaluate subchannel thermal-hydraulic conditions for both steady-state and transient conditions. The North Anna core model used an eighth-core with 12 channels for most of the DNB analyses and a more detailed model for the hydraulic analyses. The MSLB model was extended to provide detail to capture the effects of the non-uniform inlet temperature distribution across the core. An additional very detailed model was developed to analyze the effects of mixed-core configurations.

The primary differences between the Advanced Mark-BW fuel design and the resident Westinghouse NAIF fuel design are that the first grid on the Advanced Mark-BW fuel assembly is a non-mixing grid, the upper guide thimbles have slightly different diameters, and the Advanced Mark-BW fuel design includes the use of MSMGs. Thermal-hydraulically, these differences do not require different analysis techniques, with the exception of the hydraulic form loss coefficients. Therefore, the use of the LYNXT code for analyzing the performance of the Advanced Mark-BW fuel is acceptable when the appropriate form loss coefficients are included in the model.

3.6.4.2.2 DNB Correlations

Two approved critical heat flux (CHF) correlations are used in the analysis of DNB occurrence for the Advanced Mark-BW fuel. Refer to Babcock and Wilcox Topical Report BAW-10199-P-A, "The BWU Critical Heat Flux Correlations," dated December 16, 1994, and Framatome Topical Report BAW-10199-P-A, Addendum 2, "Application of the BWU-Z CHF Correlation to the Mark-BW17 Fuel Design with Mid-Span Mixing Grids," dated September 5, 2002. These are the BWU-N and BWU-Z CHF correlations. The BWU-N correlation is used for non-mixing vane grids while the BWU-Z correlation is used with enhanced mixing vane grids. These correlations address the types of mixing vane grids used in the Advanced Mark-BW fuel design. Furthermore, the DNB database, upon which the BWU-N and BWU-Z CHF correlations are based, supports using these correlations for the analyses of the Advanced Mark-BW fuel design.

In addition, the BWU-I CHF correlation is approved for mixing vane grid designs such as the mixing vanes of the NAIF fuel. Since the database for the BWU-I and BWU-N correlations include fuel designs that are very similar to the NAIF design, the use of these correlations for evaluating the NAIF fuel is acceptable. Therefore, it has been demonstrated that the BWU-I, BWU-N, and BWU-Z CHF correlations are acceptable for use in the analyses of the North Anna reactor core.

3.6.4.2.3 Coefficients

The grid form loss coefficients for the Advanced Mark-BW fuel design were developed from flow tests in the HERMES P loop under PWR primary coolant conditions. Test data and analytical information were used for the NAIF grid form loss coefficient. These form loss coefficients are used in LYNXT to model the flow behavior for DNB ratio (DNBR) and pressure drop, hydraulic loads, and crossflow velocity calculations. As part of the NRC staff audit of this proposed license amendment, the NRC staff reviewed the evaluations to determine the grid form loss coefficients and agrees that they were developed in a robust manner. Since these form loss coefficients are measured for the Advanced Mark-BW design and are created in a robust analytical manner for the NAIF assemblies, the use of these form loss coefficients for the North Anna core is acceptable.

The mixing coefficient was determined for the Mark-BW fuel design, which uses many of the same assembly components as the Advanced Mark-BW fuel design. The mixing coefficient was determined through the use of Laser Doppler Velocimeter testing for the Mark-BW fuel design. It was developed from three-dimensional velocity profiles downstream of an intermediate spacer grid over a length representing the distance between spacer grids on an assembly. Since the Advanced Mark-BW design uses spacer grids that are similar to the Mark-BW assembly design and the use of MSMGs will increase the turbulence between the spacer grids, the use of the previously developed mixing coefficient for the Advanced Mark-BW fuel design is conservative.

3.6.4.2.4 Engineering Hot Channel Factors

The engineering hot channel factors are used to account for the effects of manufacturing variations on the maximum linear heat generation rate and enthalpy rise. The local heat flux engineering hot channel factor, used in the evaluation of the maximum linear heat generation rate, is determined by statistically combining the manufacturing variances for pellet enrichment and weight. At the 95-percent probability level with a 95-percent confidence level, the value was determined to be 1.03.

The average pin power factor accounts for the effects of variations in fuel stack weight, enrichment, fuel rod diameter, and pin pitch on hot pin average power. Refer to Babcock and Wilcox Topical Report BAW-10163-P-A, Revision 0, "Core Operating Limits Methodology for Westinghouse Designed PWRs," dated June 2, 1989. The value determined for this factor is 1.03. This value is combined with other uncertainties to establish the statistical design limit (SDL).

The overall peaking factor uncertainty is a statistically combined factor that includes the effect of nuclear calculational uncertainty, local engineering hot channel factor for fuel, local engineering hot channel factor for lumped burnable poison, rod bow, and assembly bow. The overall peaking uncertainty factor was determined using Framatome's approved methodology as described in Topical Report BAW-10163-P-A, Revision 0. The overall peaking factor obtained from the Framatome methodology was lower than the uncertainty factor of 1.0815 currently used for North Anna. North Anna plans to continue using the 1.0815 peaking uncertainty factor, which is conservative.

By letter dated December 9, 2003, Framatome provided an interim report to the NRC concerning a reassessment of the methodology for determining power-peaking effects of assumed assembly bow. Subsequently, by letter dated January 22, 2004, the licensee provided the results of a plant-specific evaluation of the fuel assembly bow for the North Anna core to determine the peaking factor uncertainty. The calculated total uncertainty factor, including the revised peaking penalty of 7.8 percent, was determined to be 1.0988. In its submittal dated January 22, 2004, the licensee has committed to use this value until the fuel assembly bow deviation is generically resolved, and indicated that it will then incorporate the finding of the generic resolution into its methodology. The method used to generate the total uncertainty used conservative values and applies the peaking penalty for assembly bow in a deterministic manner instead of including it as an uncertainty for inclusion in the SDL. This approach to treating the uncertainty includes an appropriate level of conservatisms to assure that the fuel assembly bow uncertainty will be bounded and is acceptable to the NRC staff.

3.6.4.2.5 Fuel Rod Bowing

The Framatome methodology for fuel rod bow was approved in Babcock and Wilcox Topical Report BAW-10147-P-A, Revision 1, "Fuel Rod Bowing in Babcock & Wilcox Fuel Designs," May 1983. The database used to support this methodology was extended in Framatome Topical Report BAW-10186-P-A, Revison 1, Supplement 1, "Mark-BW Extended Burnup," dated November 30, 2003. This database is representative of Zircaloy clad fuel. Since M5 cladding grows at a lower rate than Zircaloy cladding under irradiation conditions, the database for Zircaloy is conservative relative to the M5 performance. Therefore, use of this database for predicting the rod bow of M5 clad fuel and continuing to use the penalty generated by the Zircaloy database for M5 fuel is conservative and acceptable for use.

3.6.4.2.6 Active Fuel Stack Height

The Advanced Mark-BW fuel design uses an increased density fuel pellet. Therefore, under irradiation conditions the pellets experience less shrinkage and subsequently the overall fuel stack shrinks less. Thermal expansion of the fuel upon initial heatup is greater than the stack shrinkage with irradiation. Therefore, neglecting the stack shrinkage and assuming a nominal fuel stack height is conservative.

3.6.4.2.7 Spike Densification Peaking Factor

The spike densification peaking factor is used to account for the peaking that is caused by inter-pellet gap formation as the result of fuel densification. To prevent the formation of large gaps between pellets and to prevent cladding creep collapse, Framatome increased the fuel density and pre-pressurized the fuel rods. As shown in Topical Report BAW-10163-P-A, the analyses used to determine the effect of the inter-pellet gaps on power peaking demonstrated that the peaking factor increase due to spike densification is negligible for the Mark-BW fuel design. Since the Advanced Mark-BW design is an evolution of the Mark-BW design and the characteristics that were used to support the determination that the spike densification peaking factor is negligible remain applicable, not applying the spike densification peaking factor to Framatome Advanced Mark-BW fuel assemblies is acceptable.

3.6.4.2.8 Core Power Distributions

The reference core axial power distribution for the Advanced Mark-BW fuel design is the 1.55 cosine. This is also the power distribution that is currently in use for the resident Westinghouse NAIF fuel. Therefore, the core axial power distribution is consistently modeled for both fuel types.

The nuclear enthalpy rise hot-channel factor is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power. The current TS COLR limit is 1.49. Eventually, the licensee would like to increase the nuclear enthalpy rise hot-channel factor. However, a change to the TS was not requested with this application. Therefore, the higher nuclear enthalpy rise hot-channel factor is not approved with this application. If in the future the higher nuclear enthalpy rise hot-channel factor is desired, the licensee will need to request a TS change and submit the associated justification.

3.6.4.2.9 Core Bypass Flow

The Advanced Mark-BW fuel assemblies create a slight increase in the pressure drop across the core. This will increase the flow through alternate routes around the core. This increase in core bypass flow was analyzed and the bounding core bypass flow was determined to be less than 5 percent when core inserts were not present. Similarly, the minimum core bypass flow was determined to be greater than 3 percent when 1500 core inserts were present. A bypass flow of 5.5 percent is specified for use in the statistical DNB analysis for the Advanced

Mark-BW fuel design and 6.5 percent is specified for non-statistical DNB applications and other deterministic nuclear steam supply system (NSSS) evaluations. The minimum bypass flow of 3 percent is used for the lift force calculations. Since the values for the core bypass flow used in the previous analysis methods are more conservative than the plant-specific values, the results of the previous analyses remain valid for the use of Framatome Advanced Mark-BW fuel.

3.6.4.3 Hydraulic Compatibility

To evaluate the pressure drop, hydraulic loads, and cross flow velocities in mixed core configurations, Framatome's mixed core methodology as outlined in section 3.6.9 of this SE is used. The calculations are performed using the LYNXT code. Four core configurations were considered: a full core of Advanced Mark-BW fuel assemblies, a full core of Westinghouse NAIF assemblies, the most limiting configuration for the Framatome Advanced Mark-BW fuel assemblies, and the most limiting configuration for the NAIF fuel assemblies. The most limiting configuration for the Advanced Mark-BW fuel is a full core of Advanced Mark-BW fuel assemblies, while the most limiting configuration for the NAIF fuel is a single fuel assembly in the center of the core with the rest of the core composed of Advanced Mark-BW fuel assemblies. The core operational conditions used for the analyses included cold zero power, hot zero power, hot full power, hot overpower, normal flow, mechanical design flow, and high flow.

3.6.4.3.1 Mixed-core Nominal Pressure Drop Results

Pressure drop evaluations demonstrated that the most limiting core configurations were obtained when a full core of Advanced Mark-BW fuel assemblies is used for the Advanced Mark-BW fuel design and when one NAIF assembly is in the center of the core with the rest of the core composed of Advanced Mark-BW fuel assemblies for the NAIF fuel design. These limiting pressure drop results are used in the evaluation of the non-LOCA transients.

3.6.4.3.2 Mixed-core Hydraulic Load Results

The results of the hydraulic load calculation is used for the evaluation of the fuel assembly holddown springs that was previously discussed in section 3.6.3.3 of this SE. The maximum hydraulic loads do not include the fuel assembly weight, buoyancy forces, or spring hold down forces. The maximum hydraulic load for the Advanced Mark-BW fuel assembly was obtained from the full-core Advanced Mark-BW core configuration. The maximum hydraulic load for the NAIF fuel assembly was obtained from the one NAIF fuel assembly in the center of the core with the rest of the core composed of Advanced Mark-BW fuel assemblies configuration.

3.6.4.3.3 Mixed-core Cross Flow Velocities

The design criterion for fretting for the Advanced Mark-BW fuel design is that the span average cross flow velocities must be less than 2 ft/sec. Analyses performed with a mixed core configuration demonstrated that the maximum cross flow velocities were seen when one NAIF fuel assembly was in the center core location with the remainder of the core consisting of Advanced Mark-BW fuel assemblies. The results of the analysis of this limiting configuration demonstrated that the cross flow velocities remained below the criteria.

3.6.4.4 DNB Performance Evaluation

To demonstrate that the DNB performance of the Advanced Mark-BW fuel design is acceptable, full- and mixed-core configurations were evaluated. The full-core analysis uses the Framatome Statistical Core Design (SCD) methodology described in Babcock and Wilcox Topical Report BAW-10170-P-A, "Statistical Core Design For Mixing Vane Cores," dated December 23, 1988, while a mixed-core analysis uses the Mixed-Core Methodology outlined in section 3.6.9 of this SE. Both types of analyses use the LYNXT code. The design criterion for use with the SCD methodology is that the minimum DNBR must be equal to or greater than the thermal design limits (TDLs), which is defined in Section 3.6.4.4.2. The design criterion for use with non-SCD methods is that the minimum DNBR must be equal to or greater than the CHF correlation design limits.

3.6.4.4.1 State Points For DNB Calculations

A set of state points were developed by the licensee for use in the DNB analyses. These state points represent points on the safety limit lines, limiting axial flux shapes at several axial offsets, and state points for several transient events, including misaligned rod, loss of flow, rod withdrawal at power, locked rotor, rod urgent failure, and rod withdrawal from subcritical and steamline break. In general, the state point conditions that were defined to evaluate the Advanced Mark-BW were used to evaluate the NAIF in the transition core analysis.

3.6.4.4.2 SCD

The Framatome SCD methodology, as described in Topical Report BAW-10170-P-A, together with the LYNXT code, is used to assess the thermal margin for the Advanced Mark-BW fuel design. The SCD method is independent of fuel type and does not specify an analysis code. The methodology in Topical Report BAW-10170-P-A has been approved for use in analyzing Framatome fuel in Westinghouse-designed reactors.

The SCD approach uses a technique that statistically combines uncertainties. In this method, the uncertainties on a group of input variables are subjected to a statistical analysis, and an overall DNBR uncertainty is established. This uncertainty is used to establish a DNBR design limit known as the SDL. Margin is then added to this limit for additional flexibility and the combination results in an analysis limit called the TDL. The calculated DNBR is compared to this TDL to demonstrate that the DNB protection is acceptable.

The TDL for the North Anna mixed core was determined to be 1.70. The SDL values are 1.61, as defined in North Anna Power Station UFSAR, Revision 39, Section 4.4.1.2, “Fuel Temperature Design Basis,” with the BWU-N correlation, and 1.31 with the BWU-Z correlation. Both of these values are less than the TDL and demonstrate that thermal margin is retained with the use of the Advanced Mark-BW fuel design.

3.6.4.4.3 Full-core SCD DNB Analysis for Advanced MARK-BW

When a full core SCD DNB analysis is performed for the Advanced Mark-BW fuel, calculations demonstrate that the minimum DNBR values are greater than or equal to the TDL of 1.70.

3.6.4.4.4 Mixed-core DNB Analysis

The mixed-core DNB analysis uses the Framatome mixed-core methodology discussed in Section 3.6.9 of this SE. The DNB calculations are performed using the LYNXT code. To determine the mixed-core penalty, the DNBR results from mixed-core configurations are compared to calculations performed at identical conditions with either a full-core Advanced Mark-BW model or a full-core NAIF model. The mixed-core penalty is equal to the largest differential value.

Various core loading patterns are evaluated to determine the most limiting core configurations for the Advanced Mark-BW fuel assembly. Similarly, the most limiting core configuration was determined for the resident NAIF fuel. Using these results, mixed-core penalties were determined for the first and second batch loads for the Advanced Mark-BW fuel, and a mixed-core penalty was determined for DNB for the NAIF fuel. The calculated mixed-core penalty factors are acceptable because it considers the important parameters in the most limiting core configurations with acceptable resolution to determine the impact of transitioning from one fuel type to another.

3.6.4.4.5 Deterministic DNBR Calculation

In the deterministic DNB analyses, the uncertainties in the state point conditions and model inputs are treated explicitly in the LYNXT input instead of incorporating them into the SDL. The Minimum DNBRs (MDNBR) calculated with the deterministic method are compared to the applicable CHF correlation design limit. The deterministic DNBR calculation method is used when the state point conditions fall outside the ranges of the SCD uncertainty propagation response surface model or if the conditions fall outside the ranges of the applicable CHF correlation. The three cases determined with the deterministic method for North Anna include rod withdrawal from subcritical, MSLB/high flow, and MSLB/low flow. The most limiting core configurations were analyzed to obtain the MDNBR. The MDNBRs calculated showed acceptable margin to the CHF correlation limits.

3.6.4.5 Overpower Delta-T and Over Temperature Delta-T Reactor Trip Functions

The thermal overpower delta-T and over temperature delta-T trip functions are discussed in the North Anna UFSAR and TS. The analytical method used to derive the limiting safety-system settings for these trip functions is approved in Westinghouse Electric Topical Reports WCAP-8745-P-A and WCAP-8746-A, “Design Basis for the Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Functions,” September 1986. The bases for these trip functions are the core thermal limit lines, axial offset envelopes, and other RCS and plant parameters. The evaluations of these trip functions with the introduction of the Advanced Mark-BW fuel assembly demonstrate that the currently licensed core thermal limit lines remain bounding and the constants for the thermal overpower delta-T and over temperature delta-T continue to protect the fuel.

The analyses demonstrated that the f(delta I) reset function for the thermal over temperature trip function needed modification. This change is necessary to conform to the power distribution validation methodology in Topical Report BAW-10170-P-A for the Advanced Mark-BW fuel assemblies. The negative end of the deadband changed from -44-percent to -35-percent axial flux difference. This corresponds to a change in the safety analysis from

-47-percent to -38-percent axial flux difference. An additional change to the f(delta I) reset function alters the maximum penalty value obtained from the f(delta I) reset function for highly top skewed power distributions. This extends the range of the reset function generator to accommodate axial flux differences between -50 percent and +50 percent (currently the range is -50 percent and +28 percent).

3.6.5 Neutronic Performance

The reload design analysis will be performed for each reload cycle in accordance with the approved methodology in Virginia Electric and Power Company Topical Report VEP-FRD-42-A, Revision 1-A, “Reload Nuclear Design Methodology,” dated November 4, 1986, and Virginia Electric and Power Company Topical Report VEP-FRD-42-A, Revision 2-A, “Reload Nuclear Design Methodology,” dated August 29, 2003. This analysis will verify that all applicable design criteria will be met.

Physical differences between the two fuel types, such as the use of M5 material, is modeled explicitly for the neutronic calculation. The impact of M5 on the neutronic calculation has been shown to be successfully modeled by comparing the estimated calculation results with the measured power distributions (axial and radial) at the Framatome LTA locations in the core.

The Advanced Mark-BW fuel contains slightly more uranium than an equally enriched Westinghouse NAIF assembly because of the Advanced Mark-BW fuel’s higher nominal fuel density. This creates a smaller rate of reactivity depletion for the Advanced Mark-BW fuel. The rate of depletion is explicitly modeled in the cycle-specific neutronic calculations.

The changes to the neutronic model inputs to accommodate the Advanced Mark-BW fuel have shown that the Advanced Mark-BW fuel design can be accurately modeled by the approved reload design methodology. This has been demonstrated through the in-reactor use of LTAs and comparisons between the predicted and actual parameters.

3.6.6 Non-LOCA Safety Evaluation

The licensee evaluated the performance of the Advanced Mark-BW fuel assembly design under postulated non-LOCA accident conditions. All of the postulated non-LOCA accidents were reviewed to determine the potential impact of changing the fuel assembly design. The impacts to the accident analyses were introduced from the larger pressure drop across the fuel, the thermal properties of the fuel, and M5 cladding properties. Therefore, the accidents of concern are the loss of reactor coolant flow, the reactor coolant pump locked rotor/sheared shaft, and the control rod ejection.

Each UFSAR Chapter 15 transient was reviewed for the impact of the Advanced Mark-BW fuel assembly design on the calculated DNBR results. A discussion of the results is presented in section 3.6.6.2.1 of this SE.

The analyses contained within this section consider the effects of a small power uprate based on a reduction in the power calorimetric uncertainty. Although the analyses have been performed at a power level consistent with the lower calorimetric uncertainty, the uprate has not been requested in this submittal; thus, this SE does not approve the non-LOCA safety analysis for the higher power level. Further, this SE does not approve the results using the higher power level. If an uprate is desired, the licensee should resubmit the analyses contained in this section with the other needed documentation to support the uprate when the uprate is requested. This approval does not approve this method of analysis for a power uprate. The method will need to be submitted along with all other supporting documentation when a power uprate is requested.

3.6.6.1 Assessment of Impact upon NSSS Modeling Design Inputs

The major analytical computer code used by the licensee for non-LOCA safety analysis is the RETRAN system code described in Virginia Electric and Power Company Topical Report VEP-FRD-41-A, “Reactor System Transient Analyses Using the RETRAN Computer Code,” May 1985. Separate models were developed to represent full cores of the Advanced Mark-BW and NAIF fuel. The licensee performed evaluations of the following design inputs to identify changes that are needed to the input model: the plant conditions; the trip reactivity, particularly the control rod drop time; the core stored energy; and fuel and clad thermal properties.

3.6.6.1.1 Nominal Plant Conditions

The nominal plant conditions for the Advanced Mark-BW fuel assembly design are given both for the statistical and non-statistical DNB applications and deterministic evaluations. The differences in the values given in the topical are introduced from the inclusion or exclusion of the uncertainties inherent in the parameter.

3.6.6.1.2 Trip Reactivity - Control Rod Drop Time

The guide thimbles for the resident NAIF fuel assemblies are smaller than the guide thimbles for the Advanced Mark-BW fuel design. Drop time data from the Advanced Mark-BW fuel assembly LTAs demonstrates that the performance of the Advanced Mark-BW fuel assembly is within the data range of comparable Framatome 17x17 fuel assembly designs. Therefore, since the guide thimbles are larger than those of the resident fuel and the database demonstrates that the current control rod drop time is conservative for application to the Advanced Mark-BW fuel assembly design, the control rod drop time specified in the TS of 2.7 seconds will remain applicable for the Advanced Mark-BW fuel design. Therefore, any transient that is impacted by a change in control rod insertion will not be affected by the change to the Advanced Mark-BW fuel assembly design.

3.6.6.1.3 Clad and Fuel Thermal Properties

The clad and fuel thermal properties are included in the RETRAN model through changes to the input deck. The specific inputs that are modified include the clad and fuel thermal conductivity, the heat capacity, and the initial fuel temperature. These changes impact the core stored energy, which is discussed in the next section of this SE, and the rod ejection accident, which is discussed in section 3.6.6.3.4.

3.6.6.1.4 Core Stored Energy

The physical dimension of the fuel from both vendors is very similar; therefore, the initial energy stored in the core is a function of the fuel average temperature and the fuel heat capacity. An evaluation of the core-stored energy between the two fuel types has demonstrated that the change in stored energy in the fuel between hot zero-power conditions and hot full-power conditions for the Advanced Mark-BW fuel is less than what is assumed for the resident NAIF fuel. Therefore, transients that are impacted by a change in the core-stored energy, such as the loss of normal feedwater and loss of offsite power, will not be negatively impacted by the change in fuel assembly design. As such, the current analysis for these transients will be conservative for application of the Advanced Mark-BW fuel assembly design.

3.6.6.2 Assessment for Accidents Not Reanalyzed

The assessment of the impact upon the NSSS modeling design inputs showed that the remaining key design inputs that influence transient behavior either remain bounding or are unaffected by the introduction of the Advanced Mark-BW fuel. Therefore, the current transient analyses for these transients remain bounding. The transients that are not impacted by the change in fuel design and were not reanalyzed include Uncontrolled Rod Cluster Control Assembly (RCCA) Withdrawal from Subcritical; Uncontrolled RCCA Withdrawal at Power; RCCA Misalignment; Uncontrolled Boron Dilution; Startup of an Inactive reactor coolant Loop; Loss of External Electrical Load and/or Turbine Trip; Loss of Normal Feedwater; Loss of Offsite Power to Station Auxiliaries; Excessive Heat Removal Due to Feedwater System Malfunctions; Excessive Load Increase Incident; Accidental Depressurization of the RCS; Accidental Depressurization of the Main Steam System and Steamline Ruptures; Spurious Operation of the Safety Injection System at Power; and Major Rupture of a Main Feedwater Line.

3.6.6.2.1 Core Thermal (DNB) Effects

Core thermal behavior was evaluated for the transients that were not reanalyzed. The evaluations were performed at specific conditions that are representative of the most limiting conditions experienced during each transient. These conditions were defined from the UFSAR Chapter 15 events, and they provided information on the DNB performance sensitivity to power level, pressure and temperature, axial power shapes, elevated hot rod power, low flow, and non-statistical DNB events. These state points for the limiting conditions were used by Framatome to evaluate the DNB performance. These evaluations demonstrated that the DNBR at all the state points was greater than the DNBR design limit.

3.6.6.3 Accidents Reanalyzed

Evaluations of the Advanced Mark-BW fuel assembly design showed that it would impact the accident analyses for the Loss of Reactor Coolant Flow, Locked Rotor, and Rod Ejection accidents. Therefore, these accidents were reanalyzed to quantify the impact of the Advanced Mark-BW fuel assembly introduction into the reactor core.

3.6.6.3.1 Computer Codes and Models

Three computer codes are used to evaluate the transient response. The system response was analyzed using the RETRAN methodology of Topical Report VEP-FRD-41-A. The values of the primary system parameters obtained from RETRAN are input into thermal-hydraulic analysis codes to determine the minimum DNBR values. For the NAIF fuel assemblies, the DNBR is obtained using the COBRA computer code with the WRB-1 DNB correlation and LYNXT is used with the BWU DNB correlation to determine the DBNR for the Advanced Mark-BW fuel assemblies.

3.6.6.3.2 Loss of Reactor Coolant Flow

The Loss of Reactor Coolant Flow event is initiated from a simultaneous loss of electrical supplies to all reactor coolant pumps. If the reactor is at power when this occurs, the loss of flow results in a fast increase in the coolant temperature. For this event, the RETRAN computer code is used to determine the values of the system parameters. These parameter values are then used as input to the COBRA and LYNXT computer codes to determine the minimum DNBR experienced during the transient. The minimum DNBR experienced during the transient was determined to be greater than the design DNBR limit for all cases, and the results are acceptable.

3.6.6.3.3 Locked Rotor

The locked rotor event causes a rapid loss of forced circulation in one RCS loop. In addition to heatup in the core from loss of flow, RCS integrity may be challenged from the volumetric expansion of the fluid caused by heating of the RCS fluid. RETRAN is used to perform transient analyses for the DNB evaluation and for the RCS integrity evaluation. For the DNB analyses, the reactor parameters were used by COBRA and LYNXT to determine the minimum DNBR for the transient. In all evaluations, the minimum DNBR was greater than the DNBR limit for the transient conditions. The RETRAN evaluation of the RCS during the transient demonstrated that for the worst conditions, the peak pressure in the RCS remained below 110 percent of RCS design pressure, which is the acceptance criterion established by the ASME Code. These analyses demonstrate that the Advanced Mark-BW and NAIF fuel assembly performance during these transients met the design criteria.

3.6.6.3.4 Rod Ejection

This accident occurs when there is a mechanical failure of a control rod mechanism pressure housing that results in the ejection of an RCCA and drive shaft. The analysis for this accident used the approved methodology in Virginia Electric and Power Company Topical Report VEP-FRE-2-A, “VEPCO Evaluation of the Control Rod Ejection Transient,” December 1984. The analysis of this transient used the RETRAN computer code with the point kinetics option. Evaluations were performed for both a full core of NAIF fuel assemblies and a full core of Advanced Mark-BW fuel assemblies. The results demonstrated that for all conditions during the cycle, the criteria outlined in the UFSAR will be met with the insertion of the Advanced Mark-BW fuel.

Results were reported for evaluations performed to demonstrate that North Anna will be able to meet the RIA criteria for this event proposed by EPRI Report TR-1002865, “Topical Report on Reactivity Initiated Accident: Bases for RIA Fuel Rod Failure and Core Coolability and Criteria,” dated April 22, 2002. The EPRI criteria are currently under review and are not approved for use. Therefore, if the criteria are approved and if the licensee should desire to use those criteria, in requesting an amendment, the analysis and methodology demonstrating that they are met will need to be submitted to the NRC for review. This SE does not approve the results of the portion of the analysis pertaining to the proposed EPRI criteria.

3.6.7 Applicability of the Licensee's Reload Design Methodology

The licensee revised its reload nuclear design methodology to accommodate the use of the Framatome Advanced Mark-BW fuel assembly design. This methodology was approved in Virginia Electric and Power Company Topical Reports VEP-FRD-42-A, Revision 1-A, and VEP-FRD-42-A, Revision 2-A. Additionally, the NRC staff reviewed Virginia Electric and Power Company Topical Report VEP-NE-1-A, "Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications," dated April 21, 1986, and determined that it was not fuel-specific and, therefore, could be used with the Framatome Advanced Mark-BW fuel assembly design without revision.

3.6.8 Spent Fuel Pool Criticality

Because the Advanced Mark-BW fuel design utilizes an increased nominal fuel density as well as different fuel density and pellet volume tolerances of the spent fuel pool, criticality analysis was reviewed. The new fuel density and density and pellet volume tolerances were evaluated to verify that the current spent fuel pool TS criticality limits were bounding for Advanced Mark-BW fuel assemblies. The calculations performed with modifications to the input parameters to emulate the new fuel demonstrated that the current TS criticality limits are bounding and need not be revised to support the use of Advanced Mark-BW fuel. Based on the applicability of the current methodology and TS limits, the NRC staff finds the use of Advanced Mark-BW fuel to be acceptable for storage in the spent fuel pool.

3.6.9 Framatome Mixed-Core Analysis Methodology Description

When a new fuel design is inserted on a reload basis, thermal-hydraulic compatibility needs to be assured between the resident and new fuel assemblies. This compatibility needs to extend to protection for DNBR, hydraulic loads, and crossflow velocities. The design criteria used to ensure that compatibility exists are:

1) During Condition I and II events, there is at least a 95-percent probability with a 95-percent confidence level that the hot pin will not experience a DNB; and a 99.9-percent probability that DNB will not occur on a core-wide basis.

2) The largest hydraulic loads will be such that there is sufficient fuel holddown margin.

All of the mixed core analyses use the LYNXT computer code to model the thermal-hydraulic behavior of the system. A number of different core configurations were modeled with most of the models using the standard 12-channel one-eighth core. Additional models provide more detail in the area of interest to evaluate localized thermal-hydraulic parameters. The DNBR performance LYNXT model uses additional nodes in the radial and axial directions. The LYNXT model used for hydraulic loads and crossflow analyses models each physical fuel assembly in an eighth-core as a separate channel. All of these models use the appropriate form loss coefficients for the specific fuel type being modeled.

Four different parameters are evaluated during a transition from one fuel type to another. These are DNBR, pressure drop, hydraulic loads, and span average crossflow velocity.

For the mixed-core evaluation of the DNBR, the design criterion is assessed by comparing the calculated DNBR to critical heat flux correlation design limits. This can be done using deterministic or statistical techniques. The process of determining the DNBR includes developing a range of boundary conditions that covers the expected range of core thermal-hydraulic conditions and determining the MDNBR for each set of boundary conditions. These MDNBRs are used to determine the transition core penalty that is applied to the DNBR.

A more detailed LYNXT model is used for evaluating the pressure drop, hydraulic loads, and crossflow velocity for mixed-core analyses. The model evaluated the conditions for the limiting core configurations for each fuel type. For these analyses, the boundary conditions cover the range of reactor operation, including low temperature isothermal operation, with all the reactor coolant pumps operating. The results of these analyses for the pressure drop, hydraulic loads, and span average crossflow velocities are used in the mechanical analyses for the fuel assembly holddown margin, gripping forces on fuel assembly components, and flow-induced vibration discussed earlier in this SE.

This mixed-core methodology is acceptable for use because it considers the important parameters in the most limiting core configurations with acceptable resolution to determine the impact of transitioning from one fuel type to another.

3.6.10 Technical Specification Modifications

Implementation of the Framatome Advanced Mark-BW fuel assembly design in the North Anna reactor core necessitates modifying three TS. These modifications are changing the peak fuel centerline equation safety limit, changing the fuel assembly design description, and incorporating approved methods into the COLR references section.

The peak fuel centerline equation safety limit is currently maintained less than 4700 degrees F. This was the generic fuel centerline melt limit determined by the Atomic Energy Commission and does not include the effects of burnable poisons. The modification to the peak fuel centerline equation would incorporate the approved vendor-specific fuel centerline melt equations into the TS. Because this change would provide a more realistic limit for the fuel, and these equations have been approved for specific application to each of the vendor's fuel products, incorporation of these limits in TS 2.1 for the specific fuel type is acceptable. The peak fuel centerline equations are referenced in VEPCO's submittal dated May 9, 2003 for Framatome Advanced Mark-BW Fuel and the North Anna Power Station UFSAR, Revision 39, Section 4.4.1.2, "Fuel Temperature Design Basis," for the NAIF.

The modification to TS 4.2 would change the description of the fuel assembly to include the use of the M5 cladding material. The M5 material has received NRC approval in Topical Report BAW-10227-P-A for use in reactors and is the cladding material used for the Advanced Mark-BW fuel assembly design. Modifying the fuel assembly description to include the use of M5 is acceptable since it is an approved cladding material.

Topical Report VEP-FRD-42, Revision 1-A listed the applicable computer codes, correlations, and methods used for thermal-hydraulic analyses of reload cores at North Anna and Surry Power Stations. Topical Report VEP-FRD-42, Revision 2 no longer identifies the specific core thermal-hydraulic methods used; rather, it states that the applicable codes and correlations for thermal-hydraulic analyses are listed in the COLR section of the North Anna TS.

NRC GL 88-16, “Removal of Cycle-Specific Parameter Limits From Technical Specifications,” dated October 3, 1988, provides for prior NRC staff review and approval for all methodologies used to calculate cycle-specific parameters that are in the COLR and are referenced in the COLR TS section. The thermal-hydraulic methodologies the licensee currently applies for North Anna Power Station (the WRB-1 DNB correlation, the licensee’s COBRA code, and a statistical design methodology) are approved for use with the current Westinghouse fuel loaded in the North Anna core. Thermal-hydraulic methodologies used in designing reload cores are typically fuel specific. Thus, in accordance with Topical Report VEP-FRD-42, Revision 2 methodology, in order to use Framatome ANP Advanced Mark-BW fuel, the licensee seeks to add the applicable and approved thermal-hydraulic methodology references to the COLR TS section. The methodologies the licensee proposes to add to the COLR TS have been approved by the NRC staff, and the conditions for their use are satisfied at North Anna, Unit 2. Therefore, the NRC staff finds that generic reference to the thermal-hydraulic methodology listed in the COLR TS section is acceptable.

3.6.11 Conclusions

The NRC staff has reviewed the licensee’s submittal and supporting documentation. Based on the considerations above, the NRC staff has concluded that the use of the Framatome ANP Advanced Mark-BW fuel assembly design and the proposed TS changes to implement the Advanced Mark-BW fuel design are acceptable because the licensee demonstrated that all applicable design criteria and regulatory requirements are met with the use of the Advanced Mark-BW fuel assembly design in the North Anna reactor core.

Additionally, the application requested review of the thermo-mechanical design of the Framatome Advanced Mark-BW fuel, as proposed for use at North Anna, Unit 2, up to a peak rod burnup of 60,000 MWD/MTU, and the NRC staff considered this burnup in its evaluation. Accordingly, as part of implementation, the amendment requires the licensee to describe, in the UFSAR, that the thermo-mechanical design of Framatome ANP Advanced Mark-BW fuel design has been evaluated up to a peak rod burnup of 60,000 MWD/MTU for use at North Anna, Unit 2, as set forth in this SE.

3.7 Spent Fuel Pool Storage and Cooling

In reviewing the proposed license amendment, the NRC staff considered the effects of the proposed Framatome ANP Advanced Mark-BW fuel on the spent fuel storage and cooling system at North Anna. In its submittal dated May 9, 2003, the licensee stated that assemblies for the Framatome ANP Advanced Mark-BW fuel are indistinguishable from the existing Westinghouse NAIF design in terms of physical appearance and interface with fuel handling equipment and spent fuel pool features such as storage racks. The licensee stated that there is no impact on the spent fuel pool structure, decay heat, thermal capacity, maximum bulk temperature, or time to boil from the use of Framatome ANP Advanced Mark-BW fuel. As such, the licensee did not request any changes to the licensing basis for the spent fuel pool or spent fuel pool cooling system. Based on the information above, the NRC staff finds the spent fuel pool storage and cooling system aspects of the license amendment request to be acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Virginia State official was notified of the proposed issuance of this amendment on March 10, 2004. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that this 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 this amendment involves no significant hazards consideration, and there has been no public comment on such finding (68 FR 43397). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

6.0 CONCLUSION

The 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. NUREG-0800, "Standard Review Plan," Section 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance," Branch Technical CSB 6-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation," dated July 1981

2. Framatome ANP Topical Report EMF-2103 (P), Revision 0, "Realistic Large Break Loss of Coolant Accident Methodology for Pressurized Water Reactors," dated September 15, 2003, Adams Accession No. ML032691418

3. Siemens Power Corporation Topical Report EMF-CC-39(P), Revision 2, "ICECON: A Computer Program Used to Calculate Containment Back Pressure for LOCA Analysis (Including Ice Condenser plants)," dated November 30, 1999, Adams Accession No. ML012880305

4. Wagner, Richard J., and Wheat, Larry L., CONTEMPT-LT Users Manual, Aerojet Nuclear Company Interim Report, I-214-74-12.1, dated August 1973

   Wheat, L. L., CONTEMPT-LT/022 Program Transmittal LLW-13-73, Letter to Argonne Code Center, Aerojet Nuclear Company, dated December 19, 1973

5. Framatome ANP Topical Report EMF-2102 (P), Revision 0, "S-RELAP5: Code Verification and Validation," dated August 31, 2001, Section 3.14, Adams Accession No. ML012880267

6. T. L. George, et al., GOTHIC Containment Analysis Package, Version 6.1, Developed by Numerical Applications, Inc, Richland Washington, for the Electric Power Research Institute (EPRI), dated July 1999

7. Letter from Leslie N. Hartz, Vice President-Nuclear Engineering, Virginia Electric and Power Company, to the USNRC, dated January 6, 2004, Adams Accession No. ML040140510

8. H. Uchida, A. Oyama, and Y. Toga, "Evaluation of Post-Incident Cooling Systems of Light Water Reactors," Proceedings of Third International Conference on the Peaceful uses of Atomic Energy, Volume 13, Session 3.9, United Nations, Geneva (1964)

9. T. Tagami, "Interim Report on Safety Assessments and Facilities Establishment Project in Japan for Period Ending June 1965 (No. 1)," Prepared for National Reactor Testing Station, dated February 28, 1966 (unpublished work)

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12. Letter from Leslie N. Hartz, Virginia Electric and Power Company to USNRC, "Virginia Electric and Power Company North Anna Power Station Units 1 and 2, Proposed Technical Specifications Changes and Exemption Request use of Framatome ANP Advanced Mark-BW Fuel," Docket Nos. 50-338/339, March 28, 2002, ADAMS Accession No. ML020930212

13. Letter from L. N. Hartz, Virginia Electric and Power Company to USNRC, "Virginia Electric and Power Company North Anna Power Station Units 1 and 2, Request for Additional Information Proposed Technical Specifications Changes and Exemption Request to use Framatome ANP Advanced Mark-BW Fuel," Docket Nos. 50-338/339, May 9, 2003, ADAMS Accession No. ML031400726

14. Letter from Leslie N. Hartz, Virginia Electric and Power Company to USNRC, "Virginia Electric and Power Company North Anna Power Station Units 1 and 2, Proposed Technical Specifications Changes and Exemption Request for use of Framatome ANP Advanced Mark-BW Fuel Request for Additional Information Regarding Assessment of Fuel Assembly Bow Concern," Docket Nos. 50-338/339, January 22, 2004, ADAMS Accession No. ML040290810

15. NUREG-0800, "Standard Review Plan," Section 4.2, "Fuel System Design," Revision 2, U.S. Nuclear Regulatory Commission, July 1981.

16. USNRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications," dated October 3, 1988.

17. Framatome Topical Report BAW-10227-P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," dated February 29, 2000, ADAMS Accession No. ML003686365.

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20. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, 1992.

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26. Framatome Topical Report BAW-10199-P-A, Addendum 2, "Application of the BWU-Z CHF Correlation to the Mark-BW17 Fuel Design with Mid-Span Mixing Grids," dated September 5, 2002

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31. Babcock and Wilcox Topical Report BAW-10170-P-A, "Statistical Core Design For Mixing Vane Cores," dated December 23, 1988.

32. Westinghouse Electric Topical Reports WCAP-8745-P-A and WCAP-8746-A, “Design Basis for the Thermal Overpower Delta-T and Thermal Over temperature Delta-T Trip Functions,” September 1986.

33. Virginia Electric and Power Company Topical Report VEP-FRD-42-A, Revision 1-A, “Reload Nuclear Design Methodology,” dated November 4, 1986.

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35. Virginia Electric and Power Company Topical Report VEP-FRD-41-A, “Reactor System Transient Analyses Using the RETRAN Computer Code,” May 1985.

36. Virginia Electric and Power Company Topical Report VEP-FRE-2-A, “VEPCO Evaluation of the Control Rod Ejection Transient,” December 1984.

37. Virginia Electric and Power Company Topical Report VEP-NE-1-A, “Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications,” dated April 21, 1986.

38. North Anna Power Station, Updated Final Safety Analysis Report, Revision 39, Section 4.4.1.2, “Fuel Temperature Design Basis”

39. Babcock and Wilcox Topical Report BAW-10168-P-A, Revision 3, “RSG LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants,” dated January 31, 1997

40. Letter from L. N. Hartz, Virginia Electric and Power Company to USNRC, “Virginia Electric and Power Company Surry Power Station Units 1 and 2, North Anna Power Station Units 1 and 2, Annual Report and 30-Day Report of Emergency Core Cooling System (ECCS) Evaluation Model Changes Pursuant to the Requirments of 10 CFR 50.46,” dated May 21, 2003, ADAMS Accession No. ML031490403

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