SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 188 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY - MINNESOTA

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

(TAC NO. MF2479)

1.0 INTRODUCTION

Application

By application dated July 15, 2013 (Reference 1), as supplemented by letters dated January 31, 2014, March 12, 2014, April 29, 2014, May 9, 2014 (two letters), and November 11, 2014 (References 2, 3, 4, 5, 42, and 6, respectively), Northern States Power Company – Minnesota (NSPM, the licensee), doing business as Xcel Energy, Inc., requested an amendment to the renewed facility operating license and the technical specifications (TSs) for the Monticello Nuclear Generating Plant (MNGP). Once approved, the amendment would allow transitioning to the AREVA ATRIUM 10XM fuel design and implementing of AREVA safety analysis methods.

The licensee requested the following changes to the MNGP TSs to reflect the use of fuel and safety analysis methods appropriate for the AREVA ATRIUM 10XM fuel bundle design:

- TS 2.1, "Safety Limits," to revise the applicability of TS 2.1.1.1 and TS 2.1.1.2 from a reactor steam dome pressure value of less than 686 psig [pounds per square inch gauge] to a reactor steam dome pressure of less than 586 psig when using AREVA methods.

- TS 4.2.1, "Fuel Assemblies," will be revised from present language, which specifies that the fuel assemblies shall consist of fuel rods and water rods, to specify instead that the fuel assemblies shall consist of fuel rods and water rods or channels.

- TS 5.6.3, "Core Operating Limits Report (COLR)," will be revised to add AREVA safety analysis methods to the references listed in TS 5.6.3.b.

The submittal included reports describing the AREVA safety analysis methods and their applicability to MNGP; the fuel design and thermal hydraulic analysis; and analyses for

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anticipated operational occurrences (AOOs) and design basis accidents (DBAs) using AREVA methods.

As the licensee described in Section 4.1 of its application (Reference 1), MNGP is not generally licensed to the current General Design Criteria (GDC) of Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, which were published in 1971. The applicable MNGP principal design criteria predate these Appendix A criteria. The MNGP principal design criteria are listed in the MNGP Updated Safety Analysis Report (USAR), Section 1.2, "Principal Design Criteria." In 1967, the Atomic Energy Commission (AEC) issued for public comment a revised set of proposed GDC (32 FR 10213, dated July 11, 1967). An evaluation comparing the MNGP design basis to the AEC-proposed GDCs of 1967 is presented in the MNGP USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria."

The associated Appendix A GDCs applicable to the NRC staff's review are further discussed in the respective sections of this safety evaluation (SE), as applicable.

The licensee's supplemental letter dated November 11, 2014, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on September 9, 2014 (79 FR 53460).

<u>Background</u>

In December 2013, the NRC approved implementation of an extended power uprate (EPU) (Reference 7) for MNGP. MNGP is currently licensed to operate at 2004 megawatts-thermal (MWt). The previously licensed thermal power level was 1775 MWt.

In March 2014, the NRC approved an LAR (Reference 8), which allows implementation of the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operating domain. Compared to the MELLLA operating domain, MELLLA+ allows for plant operation at higher power-to-flow ratios which can produce a higher steam void content in the reactor coolant water within the core region. The NRC staff's evaluation and approval of the MELLLA+ LAR assumes operation with a full core of Global Nuclear Fuels (GNF) GE14 fuel. If the AREVA ATRIUM 10XM fuel transition is approved as presently requested, MNGP would need to revert back to operation in the MELLLA operating domain unless additional NRC approval (i.e., amendment to license) is obtained.

MNGP is a General Electric (GE)-designed BWR/3 [boiling-water reactor, Type 3] reactor. The core contains 484 fuel assemblies. This is a relatively small core compared to the remainder of the domestically licensed fleet of BWRs; most domestic BWRs have cores that contain greater than 700 fuel assemblies. In addition, the core power density and peak bundle power at MNGP remain below fleet averages.

2.0 <u>EVALUATION - REACTOR SYSTEMS BRANCH</u>

The NRC staff in the Reactor Systems Branch organized its review and evaluation in a modular format, rather than present a general regulatory evaluation followed by a technical evaluation. This evaluation is organized into topical areas, each with a standalone regulatory evaluation. This was based, in part, on the limited scope of the staff's review.

The evaluation begins with a review of the licensee's fuel thermal limits. The adequacy of the licensee's thermal limits assessment (fuel thermal-hydraulic, mechanical, and nuclear design) is provided in Section 3.0 of this safety evaluation (SE); however, the thermal limits are addressed here insofar as they establish the initial conditions and acceptance criteria for the accident and AOO analyses. For example, the operating limit minimum critical power ratio (OLMCPR) is the initial condition for the AOO and accident analyses. One acceptance criterion for AOOs is that an AOO must not result in the fuel exceeding the safety limit MCPR (SLMCPR). This acceptance criterion ensures that 99.9 percent of the fuel rods in the core avoid boiling transition, consistent with the review guidance contained in Standard Review Plan (SRP) (Reference 13), Chapter 4.4, "Thermal and Hydraulic Design."[1] The section on thermal limits is primarily intended as background information.

Following the fuel thermal limits review in Section 2.1 of this SE, the NRC staff evaluated safety analyses in three categories: (1) special events, including the American Society of Mechanical Engineers (ASME) and the anticipated transients without scram (ATWS) overpressure analyses; (2) DBAs, including the emergency core cooling system (ECCS)/loss of coolant accident (LOCA) and control rod drop accident (CRDA) analyses; and (3) the AOO and stability analyses. Stability and AOO analyses are reviewed together because they are both intended to verify that a given operating limit MCPR (OLMCPR) provides adequate margin for transient occurrences to protect exceeding the SLMCPR (see discussion about the OLMCPR in Section 2.1.1).

2.1 Thermal Limits

The thermal limits include the operating limit and safety limit (OL and SL, respectively) MCPRs, and the average planar linear heat generation rate (APLHGR).

2.1.1 Minimum Critical Power Ratio

The MCPR limits are specified to protect the fuel cladding integrity in accordance with GDC 10 from 10 CFR, Part 50, Appendix A.

Safety Limit Minimum Critical Power Ratio

The core MCPR must remain at or above the SLMCPR during steady-state operation and during AOOs. The SLMCPR includes a margin for uncertainties in plant operating parameters such as the power distribution, nuclear instrumentation, and the critical power correlation. The SLMCPR is determined using an NRC-approved, statistical process to convolute the various uncertainties as described in ANP-10307PA, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors" (Reference 25). The representative reload safety analysis, as described in Chapter 4 of ANP-3213(P), "Monticello Fuel Transition Cycle 28 Reload Licensing Analysis (EPU/MELLLA)" (located in Reference 1), supports a SLMCPR value of 1.12; however, the licensee will retain its current TS SLMCPR of 1.15. Since the TS SLMCPR is higher, or more conservative, than the calculated value, the NRC staff determined that the licensee's SLMCPR analysis supports the TS value of 1.15.

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[1] U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, Chapter 4.4, Revision 2, "Thermal and Hydraulic Design," dated March 2007 (ADAMS Accession No. ML070550060).

While the SLMCPR is analyzed and confirmed on a cycle-specific basis in accordance with NRC-approved methods, and the implementation of the SLMCPR analytic methods is addressed in a separate input, the NRC staff identified an issue with the radial power uncertainty used in the SLMCPR analysis. The radial power uncertainty includes components that address the calibration of the local power range monitor (LPRM).

It states in ANP-3213(P), "Monticello Fuel Transition Cycle 28 Reload Licensing Analysis (EPU/MELLLA)," Section 4.2, "Safety Limit MCPR Analysis," that "The radial power uncertainty used in the analysis includes the effects of up to 1 traversing incore probe (TIP) machine out-of-service or the equivalent number of TIP channels and/or up to 50 percent of the LPRMs out-of-service and a 1200 effective full-power hour (EFPH) LPRM calibration interval." Currently, MNGP Surveillance Requirement (SR) 3.3.1.1.6 requires an LPRM calibration be performed every 1000 megawatt days per ton (a separate measure of exposure related to fuel burnup rather than core power, but roughly equivalent to 1115 EFPH). In addition, SR 3.0.2 states, in part, that "The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the frequency is met." If the surveillance is performed on the maximum interval permitted by SR 3.0.2, the LPRM calibration interval may exceed that assumed in the SLMCPR analysis. The validity of the power distribution uncertainty would, therefore, be in question.

In response to RAI SRXB-4.b (Reference 4), the licensee discussed the results of supplemental SLMCPR calculations performed to investigate the effect of potentially increased power distribution uncertainty associated with an extended LPRM calibration interval pursuant to SR 3.0.2. When the SLMCPR analysis was repeated with increased uncertainties, the final SLMCPR was determined to be insensitive to the added uncertainty. Therefore, the licensee concluded that the power distribution uncertainties assumed in the SLMCPR, along with the final results of the analysis, were adequate to address the SR 3.0.2-permitted, extended LPRM calibration interval.

Based on the licensee's conclusion regarding the insensitivity of the SLMCPR to increased power distribution uncertainty, the NRC staff concludes that the licensee has addressed the concern associated with the LPRM calibration interval.

Operating Limit Minimum Critical Power Ratio

The OLMCPR applies an additional margin to the SLMCPR for AOOs. In other words, if the core MCPR remains above the OLMCPR during steady-state operation, an AOO could occur and, throughout the most severe AOO, the core-wide MCPR would not fall below the SLMCPR. The OLMCPR is determined on a cycle-specific basis using the NRC-approved suite of AREVA BWR safety analysis methods. The licensee provided the results of its AOO analyses for Cycle 28 to demonstrate adequate core design to support the AREVA fuel transition; the results of the NRC staff's review are provided in Section 2.4 of this SE. Stability analyses, which are also addressed in Section 2.4 of this SE, then provide a range of oscillation power range monitor (OPRM) setpoints as a function of the OLMCPR, in order to ensure that there is adequate protection from thermal-hydraulic instability at the chosen OLMCPR.

2.1.2 Average Planar Linear Heat Generation Rate

A maximum average planar linear heat generation rate (MAPLHGR) is applied to ensure that the fuel is not operated in a condition that would cause it to exceed the bounds of the ECCS evaluation. While the ECCS evaluation itself is performed for MNGP using a generic ATRIUM 10XM neutronics design at beginning of life conditions (as documented in ANP-3211P, "Monticello LOCA Break Spectrum Analysis for ATRIUM 10XM Fuel" (in Reference 1) and evaluated in Section 2.3 of this SE), cycle specific MAPLHGR analyses are performed using the initial fluid conditions from the ECCS evaluation, but with cycle specific core neutronics parameters. The MAPLHGR analysis assures that the core design conforms to the 10 CFR 50.46(b) acceptance criteria. The licensee provided ANP-3212P, "Monticello LOCA-ECCS MAPLHGR Limits for ATRIUM 10XM Fuel" (in Reference 1) for NRC staff review.

The MAPLHGR limits analysis is an element of the cycle-specific reload safety analysis. As such, the NRC staff bases its findings and conclusions with respect to ECCS evaluation on the LOCA Break Spectrum Analysis. The MAPLHGR limits analysis captures the cycle-to-cycle variation in the predicted peak cladding temperature (PCT) and oxidation results. For example, in Cycle 28 the analyzed MAPLHGR limit produced a predicted PCT of 2088 degrees Fahrenheit (°F), whereas the ECCS evaluation documented in ANP-3211P indicated a predicted PCT of 2130 °F. Insofar as the MAPLHGR limits analysis demonstrated conformance to 10 CFR 50.46(b) acceptance criteria for a specific cycle design, the NRC staff determined that the analysis was acceptable. The detailed review of the break spectrum analysis against the applicable regulatory requirements is contained in Section 2.3.1.

2.2. Special Events

Special events are addressed in Section 7.0, "Special Analyses," of ANP-3213(P), "Monticello Fuel Transition Cycle 28 Reload Licensing Analysis" (in Reference 1), and includes the ASME overpressure analysis and ATWS pressurization analyses. This section also includes an evaluation of core safety limits based on the pressure regulator failed open event, which is used to justify a reduction in the safety limit that specifies the reactor steam dome pressure at which the thermal power must be less than 25-percent rated. This section also discusses the fire protection analysis required pursuant to 10 CFR 50 Appendix R, as well as an evaluation of the standby liquid control system (SLCS).

ASME Overpressure Analysis

Regulatory Evaluation

Appendix A of 10 CFR Part 50, GDC-31, "Fracture prevention of reactor coolant pressure boundary," specifies, in part, that the reactor coolant pressure boundary (RCPB) be designed with sufficient margin to assure that it behaves in a non-brittle manner and that the probability of rapidly propagating fracture is minimized. The current GDC-31 is comparable to AEC-proposed GDCs 33, 34, and 35.

Overpressure protection for the reactor coolant pressure boundary (RCPB) during power operation is provided by safety relief valves on the main steamlines that discharge to the suppression pool. The NRC's acceptance criteria are based on (1) the current GDC 15 (comparable to AEC-proposed GDC 9, as further described in USAR Section 14.4), insofar as it requires that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of operation, including AOOs; and (2) the AEC-proposed GDCs 33, 34, and 35, insofar as they require the RCPB be designed with sufficient margin to assure that it behaves in a non-brittle manner and that the probability of a rapidly propagating fracture is minimized. Specific review criteria are contained in Section 5.2.2 [2] of the SRP (Reference 13).

Technical Evaluation

A reactor overpressure condition could result from a load rejection or similar event in the steam and power conversion system, a spurious main steam isolation valve closure, or a malfunction in control systems causing feedwater supply or recirculation flow to exceed steam demand. The overpressure protection system and the reactor protection system (RPS) mitigate the adverse effects of such events. Overpressure protection at MNGP is discussed in USAR Section 4.4. The system includes eight safety/relief valves (SRVs).

The licensee evaluated the effect of the fuel and methods transition on ASME overpressure protection as discussed in Section 7.1 of ANP-3213(P) (in Reference 1). The evaluation considered several initiating events, including a main steamline isolation valve (MSIV) closure, a turbine stop valve closure, and a turbine control valve closure. The analysis also considered both full-power extents of the operating domain defined at EPU MELLLA conditions.

The licensee determined that the MSIV closure was the limiting event, based on analysis using the NRC-approved COTRANSA2 plant simulator code (Reference 28). The results indicate that the MSIV closure event resulted in a peak vessel bottom pressure of 1360 psig, with 15 psi (pounds per square inch) of margin to the lower vessel pressure limit. The predicted maximum steam dome pressure was 1326 psig, with 6 psi of margin to the steam dome pressure limit.

The licensee performed the ASME overpressure analyses assuming conservative initial conditions: (1) power was assumed to be 102 percent of licensed thermal power; (2) the direct scram on MSIV position was assumed to fail; (3) the turbine bypass system was not credited; (4) 3 SRVs were assumed out of service; and (5) the maximum allowable initial dome pressure was assumed.

The NRC staff requested the licensee to justify the assumption for the maximum allowable initial dome pressure. At a lower pressure condition at the same power level, the initial steady state void fraction could be higher, leading to a greater void collapse and resultant neutron flux spike. In its letter dated January 31, 2014 (Reference 2), the licensee acknowledged the potentially limiting characteristics of a lower initial dome pressure, and confirmed that the higher pressure initial condition was more limiting. The licensee stated the following:

> …a lower initial dome pressure may experience a larger pressure increase (peak pressure – initial pressure) during the event. However, a lower initial dome pressure also has more margin to the pressure limit. AREVA calculations have shown the increase in the pressure rise during the event does not offset the increase in initial pressure margin.

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[2] U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, Chapter 5.2.2, Revision 3, "Overpressure Protection," dated March 2007 (ADAMS Accession No. ML070540076).

The licensee also provided results of an analysis, applicable to MNGP, that evaluated both initial pressure conditions, and indicated that the lower initial pressure result was bounded by the higher initial pressure result by a margin of 5 psi. The staff verified the licensee's response, which is based on AREVA's prior modeling experience, by reviewing the topical report suite describing these modeling approaches. The NRC staff was unable to locate in its record system a clear disposition for this initial condition that verified the licensee's assertion that this analysis would be applicable to MNGP. In light of the facts that the difference in peak pressures in the sensitivity analyses was 5 psi, and the licensee's indicated margin to the dome pressure safety limit was 6 psi, the NRC staff determined that supplemental information would be required to verify the applicability of the experiential analyses to MNGP. The licensee provided supplemental information (Reference 5) based on a MNGP-specific sensitivity study of initial dome pressure which confirmed that the higher pressure assumed in the original analysis produced the most limiting result. Since the licensee confirmed the most limiting initial pressure condition with a sensitivity study, the NRC staff determined that the licensee's initial condition for reactor pressure was acceptable.

The licensee also stated that the results of the ASME overpressure analyses include various adders, totaling 9 psi, to account for void-quality correlations, Doppler void effects, and thermal conductivity degradation. These adders were determined by single-effect sensitivity studies (Appendix E, ANP-3224(P), "Applicability of AREVA NP BWR Methods to Monticello," in Enclosure 6 to Reference 1). The licensee referred to previous analyses submitted by AREVA to the NRC (Reference 29) which justified that separate consideration of the effects of each non-conservatism resulted in a conservative estimation of the integral correction.

Since the previous analysis was for a representative plant, the NRC requested the licensee provide additional information to demonstrate that the representative study was applicable to MNGP. In its letter dated January 31, 2014 (Reference 2), the licensee compared key parameters from the representative plant analyses to show that the sequence of events was the same for both plants. This information provided indication that the transients were sufficiently similar to the phenomena observed in the representative analysis and could reasonably be expected to apply to MNGP.

Based on this consideration, the NRC staff determined that the licensee's assignment of a 9 psi adder was appropriate for the MNGP analyses. The 9 psi adder assigned to MNGP, determined by studying the sensitivity of the non-conservative models separately, is conservative relative to an integrated analysis that accounts for all three effects.

Conclusion

The NRC staff's review verified the following criteria in accordance with SRP 5.2.2:

- The analysis was performed using acceptable analytic methods. As discussed above, the licensee used the NRC-approved COTRANSA2 computer code, and applied additional, conservative adders to the final pressure result to account for non-conservative models that had been identified subsequent to the code's approval. Since the analytic model is NRC-approved, and modified as necessary to increase the conservatism of the results, the staff determined that the analytic methods were acceptable.

- The minimum plant configuration permitted by TS was analyzed. The licensee provided a list of its modeling assumptions. The NRC staff verified that these assumptions reflected the most conservative operation permitted by the MNGP TS.

- The analysis assumed the failure of the first safety-grade reactor trip, which is the direct scram on MSIV position indication.

- The predicted peak pressures exceeded neither 110-percent of the reactor vessel design pressure, nor the safety limit for reactor steam dome pressure.

Based on the considerations discussed above, the NRC staff concludes that the proposed transition to AREVA fuel and analytic methods is acceptable with respect to overpressure protection.

Anticipated Transient without Scram (ATWS) Pressurization Analysis

Regulatory Evaluation

An ATWS is defined as an AOO followed by the failure of the reactor portion of the protection system described in AEC-proposed GDCs 14 and 15. The probability of an AOO, in coincidence with multiple failures or a common mode failure, is much lower than the probability of any of the other events that are evaluated under SRP Chapter 15. Therefore, an ATWS event cannot be classified as either an AOO or a DBA.[3] As a result, the acceptance criteria for ATWS events are different from other SRP Chapter 15 events.

The regulatory requirement for ATWS events is provided in 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants."

The NRC staff reviewed the licensee's ATWS analysis to ensure that the peak vessel bottom pressure is less than the ASME Service Level C limit, conventionally accepted as 120 percent of the vessel design pressure, or 1500 psig. The review was performed using guidance contained in SRP (Reference 13), Chapter 15.8.[4]

Technical Evaluation

The licensee presented the results of its ATWS analysis in Section 7.2 of ANP-3213(P) (in Reference 1). The licensee analyzed ATWS events associated with main steamline isolation valve closure (MSIVC) and pressure regulator failure open (PRFO) to maximum demand. These events were analyzed at a 102-percent licensed thermal power condition, at both 99-percent and 105-percent core flow (i.e., over the full range of flow allowed at EPU MELLLA conditions). In each event, the RPS is assumed to fail and plant shutdown is accomplished through SLCS actuation. A more immediate power reduction occurs due to an automatic recirculation pump trip. The licensee provided information that indicated that the PRFO was the

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[3] Chapter 14.8 of the MNGP UFSAR states, "ATWS was not considered in the original design or licensing basis of the Monticello plant and was not addressed in the Final Safety Analysis Report."

[4] U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, Chapter 15.8, Revision 2, "Anticipated Transients Without Scram," dated March 2007 (ADAMS Accession No. ML070570008).

limiting ATWS event, and that the results of the limiting event, assuming a single SRV out-of-service, were less than the 1500 psi acceptance criterion.

The NRC staff verified that the licensee performed this analysis using acceptable analytic methods. In its letter dated January 31, 2014 (Reference 2), the licensee stated that the analysis used the NRC-approved COTRANSA2 plant simulator code (Reference 28).

Since the licensee explicitly addressed the limiting ATWS events by analyzing them using the NRC-approved COTRANSA2 code, and since the results of the analysis were less than the 1500 psi acceptance criterion, the NRC staff determined that the licensee's disposition for ATWS mitigation was acceptable.

Based on the above, the NRC staff finds that the licensee acceptably addressed the effects of the proposed fuel and analytic methods transition with respect to ATWS mitigation analysis.

Conclusion

The NRC staff reviewed the information submitted by the licensee related to ATWS. The staff concludes that the licensee adequately accounted for the effects of the proposed fuel transition on ATWS. The staff further concludes that the licensee demonstrated that it will continue to meet the requirements of 10 CFR 50.62 and the analysis acceptance criteria following implementation of the proposed fuel transition. Therefore, the staff finds the proposed fuel transition acceptable with respect to ATWS.

2.2.3 Low Pressure Safety Limit Pressure Regulator Failed Open

Section 7.3 of ANP-3213(P) (in Reference 1) states the following:

> Technical Specification for Monticello, Section 2.1.1.1, Reactor Core Safety Limits (SL), requires that thermal power shall be ≤ 25 percent rated when the reactor steam dome pressure is < 785 psig (800 psia) or core flow is < 10 percent rated. In Reference 35 [a GE safety communication], General Electric identified that for plants with the MSIV low-pressure isolation setpoint < 785 psig, there is a depressurization transient that will cause this safety limit to be violated. In addition, plants with an MSIV low-pressure isolation setpoint ≥ 785 psig may also experience an AOO that violates this safety limit (Monticello MSIV low-pressure setpoint is 809 psig).

To evaluate this potential safety concern, the licensee selected the pressure regulator failure – maximum demand (open – PRFO) to analyze the possibility of the reactor vessel dome pressure dropping below 785 psig while reactor power is above 25-percent rated.

Based on a review of MNGP USAR Chapter 14, the NRC staff concluded that the PRFO is not within the MNGP licensing basis. Instead, the licensee analyzes the loss of feedwater heating for MNGP. Since a loss of feedwater heating results in excessive heat removal by the secondary system, as would a PRFO, the events are similar. The NRC staff's review guidance applicable to this class of events is contained in the SRP (Reference 13), Chapters 15.1.1 – 15.1.4, "Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam

Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve."[5] The NRC staff's review of these events is ordinarily intended to ensure that the MCPR safety limit is not violated, that the reactor coolant pressure boundary remains intact, and that the RPS performs adequately to terminate a given event.

The pressure constraint on the applicability of the CPR safety limits is based on the validity of the critical power correlations. As discussed in the generic TS BASES contained in NUREG-1433, Revision 4, Volume 2 (Reference 30), the "GE critical power correlations are applicable for all critical power calculations at pressures ≥ 785 psig and core flows ≥ 10 percent of rated flow." The information contained in the generic TS BASES for AREVA fuel (referred to as Advanced Nuclear Fuel Corporation fuel), while similar, does not reflect the critical power correlations that the licensee proposes to use and is hence out of date.

As presently written, the TS leaves the possibility that a SL could be exceeded strictly because the reactor may be driven to a state outside the prescribed range of reactor pressures, but without a valid challenge to the fuel cladding integrity, for which the SL is intended to ensure protection. Therefore, the proposed TS revision will ensure that such protection remains, despite allowing for a broader range of reactor pressures. The supporting analysis demonstrates that, as the reactor pressure reduces, reactor conditions are such that there would be no challenge to the fuel cladding integrity.

General statements in ANP-3213(P) (in Reference 1) suggest that reactor conditions at the time of low pressure are acceptable. The licensee's January 31, 2014, supplemental letter (Reference 2) provided additional details about the analyses, as follows:

- The low pressure bound of the ACE correlation, used to analyze critical power behavior for the ATRIUM 10XM fuel, is 290.8 pounds per square inch absolute (psia);

- The low pressure bound of the SPCB/GE14 correlation, used to analyze critical power behavior of the co-resident fuel, is conservatively implemented at 571.4 psia;

- The limiting event, regarding the potential to violate the TS SL, is the PRFO at 60 percent rated thermal power and 44 percent rated core flow;

- As the steam flow increases due to the failed open pressure regulator, the dome pressure decreases and the moderator reactivity becomes negative, indicating that void formation reduces the core power;

- The relative heat flux follows the relative core power, decreasing in time even prior to reactor scram; and

- The thermal power drops below 25 percent rated before the pressure drops below the lower bounds of the critical power correlations (relative power is nearly zero by 5.5 seconds while the steam dome pressure never falls below 650.0 psia).

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[5] U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, Chapters 15.1.1 – 15.1.4, Revision 2, "Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve," dated March 2007 (ADAMS Accession No. ML070550005).

Based on the additional detail provided in the response letter, as supported by the PRFO analysis, the NRC staff concluded that the proposed revision to TS 2.1.1.1 was acceptable. The analysis results demonstrate that the expansion of the applicability of the SL would not result in a condition where fuel cladding integrity would be challenged due to a postulated PRFO event. Therefore, the TS continues to establish the requisite assurance, consistent with MNGP GDC 6, and that fuel cladding integrity will be maintained under conditions of normal operation and anticipated operational occurrences.

2.3 Design Basis Accidents

The NRC staff evaluated the postulated accidents discussed in ANP-3213(P), "Monticello Fuel Transition Cycle 28 Reload Licensing Analysis," Chapter 6 (in Reference 1). As discussed in the Emergency Core Cooling System Evaluation provided below, ECCS was reviewed in detail because the licensee's evaluation exhibited little margin toward the regulatory acceptance criteria. For the remainder of the design basis events, the NRC staff performed a limited-scope review to ensure that the licensee's use of AREVA fuel and analytic methods remains consistent with the MNGP licensing basis, and that the consequences of the analyzed events remain acceptable.

Emergency Core Cooling System Evaluation

The licensee proposed to implement AREVA's EXEM BWR-2000 ECCS evaluation model (EM). The NRC staff reviewed the ECCS evaluation discussed in ANP-3211(P), "Monticello EPU LOCA Break Spectrum Analysis for ATRIUM^{TM} 10XM Fuel" (in Reference 1). Since the MNGP ECCS evaluation previously indicated less than 100°F margin to the 2200°F regulatory limit for predicted PCT, and the AREVA evaluation results for ATRIUM 10XM fuel continue to indicate little margin to the acceptance criterion, the NRC staff reviewed the ECCS evaluation results in detail.

Regulatory Evaluation

Appendix A of 10 CFR Part 50, GDC-4, "Environmental and dynamic effects design basis," specifies, in part, that structures, systems and components (SSCs) important to safety be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and that such SSCs be protected against dynamic effects. The current GDC-4 is comparable to AEC-proposed GDCs 40 and 42.

Appendix A of 10 CFR Part 50, GDC-35, "Emergency core cooling," specifies, in part, that an abundant ECCS must be provided to transfer heat from the reactor core following any LOCA. The current GDC-35 is comparable to AEC-proposed GDCs 37, 42, and 44.

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

The NRC staff's review covered (1) the licensee's determination of break locations and break sizes; (2) postulated initial conditions; (3) the sequence of events; (4) the analytical model used for analyses, and calculations of the reactor power, pressure, flow, and temperature transients; (5) calculations of PCT, total oxidation of the cladding, total hydrogen generation, changes in core geometry, and long-term cooling; (6) functional and operational characteristics of the reactor protection and ECCS systems; and (7) operator actions. The NRC's acceptance criteria are based on (1) 10 CFR § 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance; (2) 10 CFR Part 50, Appendix K, insofar as it establishes required and acceptable features of evaluation models for heat removal by the ECCS after the blowdown phase of a LOCA; (3) AEC-proposed GDCs 40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects that might result from plant equipment failures, as well as the effects of a LOCA; and (4) AEC-proposed GDCs-37, 42, and 44, insofar as they require that a system to provide abundant emergency core cooling be provided so that fuel and clad damage that would interfere with the emergency core cooling function will be prevented.

Specific review criteria are contained in SRP (Reference 13), Sections 6.3[6] and 15.6.5.[7]

Technical Evaluation

The NRC staff evaluated the licensee's ECCS evaluation by:

- Comparing AREVA's results to the prior, EPU results;
- Reviewing the general performance of the break spectrum; and
- Evaluating the phenomena associated with the limiting transient

The NRC staff requested that the licensee provide additional information about the limiting single failure for the small-break LOCA (SBLOCA), and about the limiting power shape for the large break accident. The NRC staff also requested additional information regarding the PCT results depicted in Figure 6-22 of ANP-3211(P) (in Reference 1).

Comparison of Current and Prior Results

Although the GEH and AREVA methods were developed using different regulatory framework, both methods are intended to satisfy the required and acceptable features of ECCS evaluation models set forth in Appendix K to 10 CFR Part 50. A reasonable comparison of the characteristics of the limiting break and the basic break spectrum is expected, and any differences should be readily explainable. Otherwise, discrepancies between the two break spectra may indicate that a break spectrum analysis is inadequate, or that attributes of an evaluation model may be inappropriate or inapplicable to the plant and state conditions being evaluated.

[6] U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, Chapter 6.3, Revision 3, "Emergency Core Cooling System," dated March 2007 (ADAMS Accession No. ML070550068).

[7] U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, Chapter 15.6.5, Revision 3, "Loss-of-Coolant Accident Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary," dated March 2007 (ADAMS Accession No. ML070550016).

The large-break LOCA (LBLOCA) characteristics and limiting PCTs are similar (Reference 31). The NRC staff reviewed additional details concerning the break spectrum analysis results, analyzed power shapes, and limiting small break behavior. Overall, the results were reasonably consistent, and differences between the two were attributable to specific characteristics of each evaluation model. One example is discussed below.

Although the SBLOCA characteristics are slightly different, the NRC staff determined that this result would be expected because of differences in the two vendors' models that affect the predicted effectiveness of low pressure ECCS. Additional discussion of the AREVA modeling approach is provided in Section 4.4 of ANP-3211(P) (in Reference 1).

The NRC staff reviewed the proposed ECCS evaluation results by comparing them to the prior analyses of record and determined that the predicted PCTs were reasonably consistent, and that differences in the break spectrum and limiting results were acceptable, given the differences between the two evaluation models.

Break Spectrum Behavior

The range of large break loss-of-coolant accidents is reasonably consistent, with a general trend among the split breaks to decrease steadily as a function of break size, for breaks less than the limiting size. A similar trend can be observed for breaks larger than the limiting break size. The mid-peaked and top-peaked results show very little difference; this trend can be expected for large breaks, since the rapid blowdown and faster core recovery diminish the significance of the peak power elevation. Since the licensee's results for the larger end of the break spectrum are consistent with expected performance, the NRC staff determined that the results are acceptable.

For the smaller break sizes, the NRC staff requested that the licensee provide additional detail about the single failure analysis. The limiting single failure was determined to be a LBLOCA with the failure of a low-pressure coolant injection (LPCI) injection valve. The failure of a LPCI system may not be the most limiting for smaller breaks, where low pressure injection systems would not provide the most significant sources of emergency core coolant early in the transient. Thus, the NRC staff requested that the licensee provide results for the limiting small break single failure.

In response to SRXB RAI-2 (Reference 2), the licensee provided the results for the SBLOCA break spectrum with the assumed station battery failure. The spectrum illustrated generally limiting behavior for all locations and power shapes at the limiting break area, and the limiting power shape was top-peaked.

Based on the above, the NRC staff finds that since a SBLOCA is characterized by a slower boiling down of the core liquid and a slow recovery of the limiting elevation, the top-peaked limiting result is expected and hence acceptable.

The Limiting Large Break

The NRC staff evaluated the limiting large break behavior depicted in Figures 6.1 through 6.22 of ANP-3211(P) (in Reference 1). The figures were generally consistent with the generic large break behavior described in Chapter 3 of ANP-3211(P). Figure 6.22, "Limiting TLO Recirculation Line Break Cladding Temperatures," depicted a brief temperature excursion for the PCT rod between 125-150 seconds. In response to SRXB RAI-1 (Reference 2), the licensee clarified that this "blip" occurred because of a predicted cladding rupture on the PCT rod at the plane of interest. This response indicated that the figure was depicting predicted phenomena, and confirmed that the result was valid.

Based on the consistency of the limiting break behavior with the general behavior expected in a LBLOCA event, and on the licensee's clarification that the PCT trend exhibited expected behavior (at cladding temperatures in excess of 1900°F, cladding rupture is expected), the NRC staff finds that the results for the limiting break are acceptable.

Results from the Limiting Event

From the limiting LBLOCA, the following results were provided in Table 6.1 of ANP-3211(NP) and corrected, in part, by Reference 6. According to Table 6.1, the PCT is 2130°F, the maximum local cladding oxidation is 3.96 percent, and the maximum planar average metal water reaction is 1.12 percent; corrected to 1.17 percent in Reference 6. The acceptance criteria established in 10 CFR 50.46(b)(1) – (3) require PCT to be less than 2200 °F, maximum local oxidation to be less than 17 percent, and hydrogen generation to be less than an equivalent to 1 percent of total cladding reacted in the core. Although 1.17 percent may appear to exceed the hydrogen generation limit at 10 CFR 50.46(b)(3), the axial peaking factor at the limiting location exceeds 1.5, meaning that the total equivalent cladding reacted would be far less than 1 percent, and the result is thus acceptable.

The acceptance criteria contained in 10 CFR 50.46(b)(4) and (b)(5) require that the fuel cladding remain in a geometry amenable to cooling, and that adequate cooling be provided for the long-term removal of decay heat generated by the core. Chapter 8 of ANP-3211(P) (in Reference 1) disposes these acceptance criteria by reference to the design characteristics inherent in a BWR. Specifically, provided that ECCS liquid can be provided to cover two-thirds of reactor core height, a stable quench can be maintained and the top third of the core is adequately cooled with core spray. This disposition is based on the BWR system design and unrelated to the fuel design or analytic methods. The NRC staff finds that the disposition is acceptable on this basis.

Conclusion

Based on the NRC staff evaluation summarized above, the NRC staff determined that the proposed transition to AREVA fuel and safety analysis methods is acceptable with respect to the ECCS evaluation. The licensee proposes to implement an NRC-approved evaluation model, in conformance with 10 CFR 50 Appendix K. In turn, this implementation also conforms to 10 CFR 50.46(a)(1)(ii), which states, "… an ECCS evaluation model may be developed in conformance with the required and acceptable features of appendix K ECCS evaluation models." The results provided by the licensee ensure that ECCS performance has been "calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents" have been calculated, as required by 10 CFR 50.46(a)(1)(i). The licensee's results showed that the limiting LBLOCA remained within 10 CFR 50.46(b) acceptance criteria.

Based on these considerations, the NRC staff concludes that the ECCS evaluation, its results, and the proposed implementation of the EXEM-BWR ECCS evaluation model are acceptable, and that the proposed fuel design change is also acceptable.

2.3.2 Non-LOCA Postulated Accidents

The licensee described the effect that the fuel and safety analysis methods transition would have on the plant's predicted performance for the remaining, non-LOCA postulated accidents within the MNGP licensing basis. This included the following:

- Recirculation Pump Seizure Accident
- Control Rod Drop Accident
- Fuel and Equipment Handling Accident
- Fuel Loading Error

Recirculation Pump Seizure Accident

The licensee evaluated a recirculation pump seizure event from both two-loop and single-loop operating conditions. The pump seizure from two-loop operation is characterized as an event of minor consequence. The pump seizure event from single-loop operation (SLO) has the potential to be limiting; however, the licensee's process will modify the MCPR operating limits for SLO, if necessary, to assure this accident does not violate the AOO acceptance criteria. In its review, the NRC staff considered (1) the licensee's disposition for this event is common for BWRs, and (2) the effects of the event are sensitive to core power and recirculation loop flow conditions, more so than specific bundle design characteristics.

Based on the above considerations, the NRC staff determined that the licensee's disposition for the recirculation pump seizure accident was acceptable insofar as it supports the proposed transition to AREVA fuel and safety analysis methods.

Control Rod Drop Accident

Regulatory Evaluation

The NRC staff evaluated the consequences of a CRDA in the area of reactor physics. The staff's review covered the occurrences leading to the accident, the safety features designed to limit the amount of reactivity available and the rate at which reactivity can be added to the core, the analytical model used for analyses, and the results of the analyses. The NRC's acceptance criteria are based on GDC-28 (comparable to AEC-proposed GDC-32), insofar as it requires that reactivity control systems are designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core. The specific review criteria are contained in SRP (Reference 13), Section 15.4.9.[8]

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[8] U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, Chapter 15.4.9, Revision 3, "Spectrum of Rod Drop Accidents," dated March 2007 (ADAMS Accession No. ML070550015).

Technical Evaluation

The licensee performed an MNGP-specific CRDA analysis using NRC-approved AREVA analytic methods described in XN-NF-80-19(P)(A), Volume 1 and Supplements 1 and 2 (Reference 32). The analysis considered both co-resident GE14 and ATRIUM 10XM fuel designs, and was performed using the CASMO-4/MICROBURN-B2 code system documented in EMF-2158(P)(A) (Reference 24).

The NRC staff applied two acceptance criteria to the results of the CRDA: the maximum deposited fuel enthalpy should be less than 230 calories per gram (cal/g) to assure core coolability, and an applied 170 cal/g fuel damage threshold for the purpose of determining the number of rods with cladding failure for the radiological consequences. The licensee confirmed that the maximum deposited fuel enthalpy was 228 cal/g, and the number of ATRIUM 10XM rods exceeding 170 cal/g was 736. The number of failed rods is less than the number of failed rods assumed in the USAR. As such, the radiological consequences remain bounded by the existing USAR analysis.

Conclusion

The licensee applied NRC-approved analytical methods and determined that core coolability and cladding failure criteria remain satisfied. Based on the above, the NRC staff concludes that the proposed fuel transition was acceptable with respect to the CRDA.

Fuel and Equipment Handling Accident

The licensee stated that the fuel handling accident (FHA) radiological analysis of record for the MNGP alternate source term was addressed with consideration of ATRIUM 10XM core source terms and number of failed fuel rods in its license amendment request dated October 30, 2012, which was approved in a letter dated October 24, 2014 (Reference 33).

Since the FHA is addressed by a separate licensing action, the NRC staff determined that a review of the fuel and equipment handling accident was not within the scope of the fuel and safety analysis methods transition review.

Fuel Loading Error

Within the MNGP licensing basis, fuel loading errors are classified as infrequent events. The licensee uses an alternative source term and ensures, as an acceptance criterion, that offsite dose associated with a fuel loading error shall not exceed a small fraction of 10 CFR 50.67 limits. AREVA analytic methods were used to evaluate both a mislocated fuel bundle and a misoriented fuel bundle. For both fuel loading errors, there was no challenge to fuel centerline melt or cladding strain limits, and the change in CPR for either event is well below the CPR change reported for limiting AOOs. Therefore, the licensee concluded that fuel cladding would remain intact and there would be no fuel melt, such that the radiological consequences evaluated in the licensing basis remain bounding for either event.

Based on the results of the licensee's analyses, the NRC staff concluded that the proposed fuel and analytic methods transition was acceptable with respect to the fuel loading error.

2.4 Anticipated Operational Occurrences and Stability

Regulatory Evaluation

The NRC staff reviewed Section 4.3, "Core Hydrodynamic Stability," and Section 5.0, "Anticipated Occurrences," of ANP-3213(P) (in Reference 1) to evaluate the applicability of the AREVA safety analysis methods to MNGP, to confirm that the use of the methods is within the NRC approved ranges, and to verify that the results of the analyses are in compliance with the requirements of the applicable GDCs contained in 10 CFR 50, Appendix A. The NRC staff's review was based on (1) GDC 10 (comparable to AEC-proposed GDC 6, as further described in the MNGP USAR Section 14.4), insofar as it requires that the RCS be designed with appropriate margin to ensure that Specified Acceptable Fuel Design Limits (SAFDLs) are not exceeded during normal operations, including AOOs, and (2) GDC 12 (comparable to AEC-proposed GDC 7, as further described in the MNGP USAR) , insofar as it requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can reliably and readily be detected and suppressed.

Technical Evaluation

Anticipated Operational Occurrences

The plant response to the limiting AOOs is analyzed for each reload cycle. As previously discussed in Section 2.1.1 of this SE, the results are used to establish the OLMCPR. To support the proposed fuel and safety analysis methods transition, the licensee provided the results of its reload transient analysis which covers the projected operating conditions within the licensed power-to-flow map, equipment out of service options, and SCRAM speed options (i.e., both the TS-required scram speed and nominal scram speed). For the initial application of AREVA fuel and methodology to MNGP, the reload analysis provided a simulation of transient events to cover rated and off-rated operating conditions. The results were used to determine the OLMCPR limits for ATRIUM 10XM and co-resident GE14 fuel.

The thermal limits are determined following the NRC-approved AREVA-proprietary THERMEX thermal limits methodology (Reference 34). The methodology employs the COTRANSA2 (Reference 28), XCOBRA (Reference 34), XCOBRA-T (Reference 35), and CASMO-4/MICROBURN-B2 (Reference 24) computer codes. COTRANSA2 is the system transient simulation code; and XCOBRA and XCOBRA-T are thermal-hydraulic codes used for steady-state and transient analysis, respectively. The CASMO-4/MICROBURN-B2 code system is mainly used for neutronics; however, the code has a thermal-hydraulic analysis capability and is used, for example, in the control rod withdrawal error analysis. This methodology, in conjunction with these computer codes, is specifically approved for BWR transient analysis and thermal limits assessment. As such, the NRC staff determined that they are acceptable for application at MNGP in its MELLLA operating domain.

Based on the cycle-specific safety analysis, the OLMCPR is established as discussed in Section 2.1.1 of this SE. The AOO events analyzed to determine the OLMCPR include the following: load rejection with no bypass; turbine trip with no bypass; turbine trip with bypass; feedwater controller failure to maximum demand; an inadvertent actuation of the high pressure coolant injection system; and a control rod withdrawal error. The results, which are provided for both

ATRIUM 10XM fuel and GE14 fuel, show similar performance between the two bundle designs (transient delta (i.e., change in) CPR (ΔCPR) differs by 0.01 typically).

The NRC staff verified, by reviewing a reload safety analysis prepared for MNGP by its current fuel vendor, that these events are the same as those currently within the plant's cycle-specific reload safety analysis scope (Reference 36). The analyses also support an OLMCPR value that is consistent with previous values.

Stability

The reactor core stability evaluations provided in Reference 1 are predicated on the NRC-approved reactor stability long-term solution (LTS) Option III (Reference 37). Option III relies on an OPRM to detect stability decay ratios and trip the reactor if destabilizing power oscillations are detected. The licensee evaluated two conditions for postulated oscillations: (1) steady-state operation at 45 percent core flow, and (2) a transient associated with a two recirculation pump trip from the full-power operation state point.

The licensee used the RAMONA5-FA computer code in accordance with the NRC-approved methodology described in BAW-10255PA (Reference 38) to calculate the relative change in CPR as a function of the calculated hot channel oscillation magnitude for each of the initiating events described above. A stability-based OLMCPR is calculated using the most limiting of (1) the RAMONA5-FA-calculated change in relative ΔCPR for a given oscillation magnitude, or (2) a generic value calculated in accordance with the Option III methodology. The licensee's calculations determined that the generic value was limiting for the cycle design presented in ANP-3213(P) (in Reference 1).

Backup Stability Protection (BSP) is implemented at MNGP when the OPRM system is inoperable. The BSP includes specific requirements for operator action as well as restrictions on operation in certain regions of the power/flow map. These BSP regions are determined using the NRC-approved STAIF methodology (Reference 39). The STAIF methodology is used to define BSP regions based on criteria related to decay ratio, or the measure of growth (or decay) of power oscillations. The decay ratio is calculated based on the neutronic feedback and the thermal-hydraulic conditions at any given region on the power-to-flow map. The results of the STAIF analysis are used to define a Region I, where immediate scram is required, and a less severe Region II, where manual intervention is required to exit the power-to-flow conditions in that region. These regions occur at the high-power and low-flow range within the MELLLA operating domain.

Conclusions

Since the licensee analyzed the limiting AOOs on a cycle-specific basis to determine the OLMCPR, and because the THERMEX results indicated reasonable agreement with the results of the prior analyses, the NRC staff determined that the licensee's application of the THERMEX methodology is acceptable. Since the results indicated adequate thermal margins for the analyzed SLMCPR and supported OLMCPR values, and since the AREVA and co-resident GE14 fuel indicated similar thermal margin performance, the NRC staff also determined that the implementation of the requested transition to ATRIUM 10XM fuel design was acceptable.

In its review of the licensee's stability analyses, the NRC staff noted that the licensee is using the AREVA suite of stability analysis methods in a manner consistent with their approval. In addition, the results provided in Section 4.3 of ANP-3213(P) (in Reference 1) show that a range of OPRM setpoints were available to support a variety of assumed OLMCPRs, such that the stability solution would provide acceptable protection at an OLMCPR that is supported by the AOO analyses. Finally, the NRC staff also reviewed the licensee's stability solution within the MELLLA operating domain during the review of the Monticello EPU and found the results acceptable. Since the licensee will operate the plant using AREVA fuel within the same MELLLA operating domain, the EPU findings, and especially those related to operator actions and general LTS Option III methodology, remain applicable for MNGP when operating with ATRIUM 10XM fuel.

Based on these considerations, the NRC staff determined that the proposed transition to AREVA fuel and safety analysis methods is acceptable for MNGP with respect to the stability analyses.

2.5 Summary and Conclusions

Based on the considerations discussed in the preceding sections, the NRC staff determined that the proposed transition to AREVA fuel and safety analysis methods at MNGP was acceptable. The staff's review supported the following conclusions:

- The overpressure and ATWS analyses show that MNGP can use ATRIUM 10XM with adequate overpressure protection to protect the integrity of the reactor coolant pressure boundary.
- The licensee has acceptably implemented COTRANSA2 for analysis of the overpressure events.
- The low-pressure safety limit analysis of the pressure regulator failed open demonstrates adequate thermal margins to justify extending the pressure applicability range of TS SL 2.1.
- The licensee has acceptably implemented the EXEM-BWR/2000 ECCS evaluation model, and has demonstrated that ATRIUM 10XM fuel can be used with adequate margins to the ECCS acceptance criteria set forth in 10 CFR 50.46(b).
- Remaining DBAs are unaffected by the proposed fuel transition, since the licensee's analyses indicated that radiological consequences would remain within those previously established in the MNGP licensing basis.
- The licensee's use of the THERMEX methodology, along with its current constituent computer codes, shows that the ATRIUM 10XM fuel can perform with similar analytic margins to co-resident GE14 fuel, and with similar analytic margins as those previously demonstrated in the current vendor's safety analysis.
- The limiting thermal margin events at MNGP will continue to be analyzed on a cycle-specific basis to determine the OLMCPR.
- The licensee has demonstrated with use of LTS Option III stability solution and RAMONA5-FA that the generic thermal margin protection setpoints remain adequate

with ATRIUM 10XM fuel, but this fact will be confirmed on a cycle-specific basis, and more conservative, plant-specific setpoints will be adopted if necessary.

- BSP setpoints have been assessed and will be confirmed using the STAIF codes, but the manual operator actions required to provide adequate BSP are unchanged, and unaffected, by the change in fuel design.

Based on the above, the NRC staff determined that the licensee's proposed transition to AREVA ATRIUM 10XM fuel and safety analysis methods is acceptable with respect to the accident and transient analyses. In addition, the NRC staff determined that the proposed revision to TS 2.1, revising the low-pressure bound of the TS Safety Limit, is acceptable.

Finally, as the licensee is proposing to implement AREVA safety analysis methods that are NRC-approved and acceptable for implementation at MNGP as discussed in this SE, the NRC staff concludes that the proposed revisions to the TS COLR References list is acceptable. The staff determines that these conclusions apply to the following references being proposed for addition to the TS[9]: References 9 through 14, 17 through 19, and 23.

Proposed References 6 through 8, and 20 are addressed by the fuel mechanical design review, while proposed References 15, 16, 21, and 22 are addressed by the fuel thermal-hydraulic design review.

3.0 EVALUATION - NUCLEAR PERFORMANCE AND CODE REVIEW BRANCH

The Nuclear Performance and Code Review branch (SNPB) of the Division of Safety Systems (DSS) staff reviewed the following sections of the licensee's license amendment request:

- Enclosure 1, Evaluation of Proposed Change
- Enclosure 5, ANP-2637, "Boiling Water Reactor Licensing Methodology Compendium," Revision 4
- Enclosure 6, ANP-3224P, "Applicability of AREVA NP BWR Methods to Monticello," Revision 2
- Enclosure 8, ANP-3119P, "Mechanical Design Report for Monticello ATRIUM 10XM Fuel Assemblies," Revision 0
- Enclosure 10, ANP-3092P, "Monticello Thermal-Hydraulic Design Report for ATRIUM 10XM Fuel Assemblies," Revision 0
- Enclosure 12, ANP-3138P, "Monticello Improved K-factor model for ACE/ATRIUM 10XM Critical Power Correlation," Revision 0
- Enclosure 14, ANP-3215P, "Monticello Fuel Transition Cycle 28 Fuel Cycle Design (EPU/MELLLA)," Revision 0

[9] Refer to Enclosure 3 of Reference 1 for the licensee's list of proposed References to add to TS 5.6.3.b.

- Enclosure 16, ANP-3213P, "Monticello Fuel Transition Cycle 28 Reload Licensing Analysis (EPU/MELLLA)," Revision 1

- Enclosure 22, ANP-3221P, "Fuel Rod Thermal-Mechanical Design for Monticello ATRIUM 10XM Fuel Assemblies," Revision 0

- Enclosure 24, ANP-3139P, "Nuclear Fuel Design Report Monticello Cycle 28 ATRIUM 10XM Fuel," Revision 1

The licensee states it will transition to AREVA ATRIUM 10XM fuel design in Cycle 29, commencing in the spring of 2017. NSPM requested approval for transition to AREVA fuel at EPU conditions with MELLLA operating domain. In order to implement the proposed use of ATRIUM 10XM fuel design at MNGP, and to adopt AREVA fuel design and safety analyses methodology, several TS changes are required. The changes are summarized below.

The proposed LAR will add eighteen (18) AREVA analysis methodologies to the TS 5.6.3 list of approved methods to be used in determining core operating limits in the COLR. The current MNGP TSs includes GNF analytical methods. The additional methodologies were listed in Enclosure 3 of Reference 1, and revised per Enclosure 2 of Reference 6.

The proposed change to TS 2.1 will revise reactor core safety limits (SLs) to reduce the value of reactor steam dome pressure in TSs 2.1.1.1 and 2.1.1.2 from 686 psig to 586 psig. The reactor core safety limits are established to maintain the fuel cladding integrity and no significant fuel damage is calculated to occur if the safety limits are not exceeded.

The proposed change will insert a minor editorial change to TS 4.2.1 and replace "water rod" with "water channel" to reflect the ATRIUM 10XM fuel assembly design feature.

3.1 Regulatory Evaluation

The AREVA ATRIUM 10XM fuel design was developed using the thermal mechanical design bases and limits as outlined in Reference 12, compliance with which ensures the fuel design meets the regulatory requirements for fuel system damage, fuel failure, and fuel coolability criteria identified in the NRC's SRP (Reference 13). The SRP is intended to provide comprehensive guidance for staff review of LARs, and also establishes the regulatory requirements in evaluating the safety of light-water nuclear power plants and their plant-specific Safety Analysis Reports (SAR).

In Section 4.2, "Fuel System design", Section 4.3, "Nuclear Design", and Section 4.4, "Thermal and Hydraulic Design," of the SRP, regulatory guidance is provided for the review of fuel rod cladding materials, the fuel system, the design of the fuel assemblies and control systems, and thermal and hydraulic design of the core. In addition, the SRP provides guidance for compliance with the applicable GDCs in Appendix A to 10 CFR Part 50.
In accordance with Section 4.2 of the SRP, the NRC staff's fuel system safety review provides assurance that:

- The fuel system is not damaged as a result of normal operation and Anticipated Operational Occurrences (AOOs);

- Fuel system damage is never so severe as to prevent control rod insertion when it is required;

- The number of fuel rod failures is not underestimated for postulated accidents; and

- Coolability is always maintained.

The NRC staff will evaluate the applicability of the AREVA methodology to the MNGP TSs and changes in the SLMCPR, to confirm that the use of the methodology is within NRC-approved ranges of applicability and to verify that the results of the analyses are in compliance with the requirements of the following GDCs specified in Appendix A to 10 CFR Part 50:

- GDC-10, "Reactor design," requiring the reactor design (reactor core, reactor coolant system (RCS), control and protection systems) to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including AOOs. The current GDC-10 is comparable to AEC-proposed GDC 6, as further described in the USAR Section 14.4.

- GDC-12, "Suppression of reactor power oscillations," requiring that power oscillations that can result in conditions exceeding specified acceptable fuel design limits are not possible, or can be reliably and readily detected and suppressed. The current GDC-12 is comparable to AEC-proposed GDC 7, as further described in the USAR Section 14.6.

- GDC-15, "Reactor coolant system design," requiring the RCS and associated auxiliary, control, and protection systems to be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including AOOs. The current GDC-15 is comparable to AEC-proposed GDC 9, as further described in the USAR Section 14.4.

- GDC-20, "Protection system functions," requiring the protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety. The current GDC-20 is comparable to AEC-proposed GDCs 14 and 15, as further described in the USAR Section 14.4.

- GDC-25, "Protection system requirements for reactivity control malfunctions," requiring the protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods. The current GDC-25 is comparable to AEC-proposed GDC 31, as further described in the USAR Section 14.4.

- GDC-26, "Reactivity control system redundancy and capability," requiring two independent reactivity control systems of different design principles be provided, one of which is capable of holding the reactor subcritical under cold conditions. The current GDC-26 is comparable to AEC-proposed GDC 27, as further described in USAR Section 14.4.

- GDC-27, "Combined reactivity control system capability," requiring the reactivity control systems to be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system (ECCS), of reliably controlling reactivity changes under postulated accident conditions. The current GDC-27 is comparable to AEC-proposed GDCs 27 and 28, as further described in USAR Section 14.4.

- GDC-28, "Reactivity limits," requiring the reactivity control systems to be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. The current GDC-28 is comparable to AEC-proposed GDC 32.

- GDC-35, "Emergency core cooling," requiring a system to provide abundant emergency core cooling to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented, and (2) clad metal-water reaction is limited to negligible amounts. The current GDC-35 is comparable to AEC-proposed GDCs 37, 42, and 44.

3.2 Technical Evaluation

3.2.1 Mechanical Design of AREVA ATRIUM 10XM Fuel

Enclosure 8 of Reference 1 (ANP-3119(P)) provides the mechanical design details and fuel structural analysis results of the AREVA ATRIUM 10XM fuel design for use at the MNGP. The fuel design is comprised of a 10x10 array of fuel rods with a square internal water channel that displaces a 3x3 array of rods, with seventy-nine (79) full-length rods (FLR), and twelve (12) part-length rods (PLFRs). The active length of a PLFR is approximately one-half the length of a FLR. Use of the PLFRs is expected to improve fuel utilization in the high void upper region of the fuel bundle, to enhance the shutdown margin, to improve stability, and improve pressure drop performance. Table 2-1 of Enclosure 8 lists the fuel assembly and component description of the ATRIUM 10XM fuel design.

The ATRIUM 10XM fuel assembly consists of a lower tie plate (LTP), 91 fuel rods, nine spacer grids, a central water channel with five water channel crowns, and miscellaneous assembly hardware. The structural connection between the LTP and upper tie plate (UTP) is provided by the central water channel.

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Fuel Rods

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Fuel Channel and Components

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3.2.2 Fuel Design Evaluation

Objectives of the fuel design are to ensure that (1) the fuel assembly (system) does not fail 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; (4) fuel coolability is always maintained; (5) the mechanical design of the fuel assemblies shall be compatible with co-resident fuel and the reactor core internals; and (6) fuel assemblies shall be designed to withstand the loads from handling and shipping.

The first four objectives are discussed in Section 4.2 of the SRP, and the latter two assure the structural integrity of the fuel and compatibility with the existing reload fuel (co-resident fuel). The fuel design evaluation contains only fuel structural analyses, where the fuel rod evaluation is documented in Enclosure 22 (in Reference 1) and will be discussed later in Section 3.2.4 of this safety evaluation.

Stress, Strain, or Loading Limits on Assembly Components

The ASME Boiler and Pressure Vessel Code (ASME B&PV) (Reference 14) was used as guidance in establishing acceptable stress, deformation, and load limits for standard fuel assembly components. These limits are applied to the design and evaluation of the UTP, LTP, spacer grids, springs, and load chain components, as necessary and applicable. The fuel assembly structural component criteria under faulted conditions are based on Appendix F of the ASME B&PV Code, Section III, with some criteria derived from component tests. Outside of faulted conditions, most structural components are under the most limiting loading conditions during fuel handling.

In response to SNPB RAI Number 9, AREVA provided a summary of its stress evaluations performed to confirm the design margin and establish a baseline for adding accident loads in determination of loading limits on fuel assembly components (Reference 2). To evaluate the stresses under normal operating conditions, **[[**

]]. The maximum normal operation **[[**
]] for MNGP is then compared against the limit to ensure that an adequate margin is maintained.

Stresses under AOO and accident conditions were evaluated using the **[[**

]].

Based on the above, the NRC staff determined that the stress evaluation results, and comparison of the results to the load limits, shows that the fuel assembly structural component criteria specified in Table 3-1, in Section 3.4.4 of Enclosure 8 (in Reference 1) is satisfied.

Fatigue and Fretting Wear

Fatigue of structural components is generally low because of small number of cycles (reactor startup) or small amplitudes. Fatigue loads on zircaloy structural components remain under the fatigue life curve determined by O'Donnell and Langer (Reference 15).

Although there is no specific wear limit for fretting, a general acceptance criterion is that fuel rod failures due to grid-to-rod fretting shall not occur. **[[**

]]. Post-test inspections of the fuel assembly showed no significant wear on fuel rods in contact with spacer springs relaxed to end of life (EOL) conditions.

Based on the above, the NRC staff finds that the lack of significant wear at the spacer cell locations relaxed to EOL conditions provides further assurance that no significant fretting will occur at higher exposure levels.

Rod Bow

Rod bow is calculated using NRC-approved model described in Reference 23. The differential expansion between the fuel rods and cage structure, and lateral thermal and flux gradients, can lead to lateral creep bowing of rods in the spans between spacer grids. **[[ ]]**. The criterion for fuel rod bowing is that the bow displacement is low enough not to impact thermal margins. Post-irradiation examinations of prior AREVA fuel designs have confirmed that such rod bow has not reduced spacing between adjacent rods **[[ ]].**

Axial Irradiation Growth

Fuel assembly components such as fuel channels must maintain clearances and engagements throughout its design life. There are three specific growth calculations for the AREVA ATRIUM 10XM fuel design: (1) minimum fuel rod clearance between LTP and UTP; (2) minimum engagement of the fuel channel with the LTP seal spring; and (3) external interfaces (e.g., channel fastener springs).

Rod growth, assembly growth, and fuel channel growth are calculated using correlations derived from post-irradiation data. Additional 10x10 fuel rod data comparisons with older fuel rod configurations were performed. Assembly growth is dictated by the water channel growth. The upper and lower tolerance limits of the growth are used to obtain EOL growth values.

Assembly Liftoff

The fuel assembly shall not levitate under normal operating, AOO, or faulted conditions. Under postulated accident conditions, the fuel shall not become disengaged from the fuel support. These criteria assure control blade insertion is not impaired. For normal operating conditions, the calculated net axial force acting on the assembly due to addition of the loads from gravity, hydraulic resistance from coolant flow, difference in fluid flow entrance and exit momentum, and buoyancy will be in the downward direction, indicating no assembly liftoff. The net force calculation is performed at maximum hot channel conditions because the greater two-phase flow losses produce a higher uplift force. Mixed core conditions for assembly lift-off are considered on a cycle-specific basis, as determined by the plant and other fuel types. The ATRIUM 10XM has greater weight, longer fuel assembly engagement length, and less pressure drop than the GE14 fuel assembly. The uplift is limited to less than the axial engagement, such that the fuel assembly becomes neither laterally displaced nor blocks insertion of the control blade.

In response to an NRC staff RAI (Reference 3) requesting details on typical calculations that show margins to assembly lift-off under normal operating and faulted conditions, the licensee stated that liftoff calculations were performed for the ATRIUM 10XM under reactor conditions at MNGP to ensure fuel design criteria established in Reference 12 are met, i.e., for normal operation and AOOs, the submerged fuel assembly weight, including the channel, must be greater than the hydraulic loads, and (2) for accident (faulted) conditions, the normal hydraulic plus additional accident loads shall not cause the assembly to become disengaged from the fuel support to assure that control blade insertion is not impaired.

The calculation consisted of two parts: **[[**

]]. The fuel assembly is compared to a qualified co-resident fuel to demonstrate that the co-resident fuel bounds the ATRIUM 10XM fuel assemblies by comparing the inlet pressure drop, fuel assembly weight, and lower tie plate collar engagement with the fuel support. The results from these calculations are listed in Table 4 of Enclosure 8 (in Reference 1), and indicate that the net force on the assembly is downward and prevents the assembly from liftoff during normal operating conditions, AOOs, and accident conditions.

3.2.3 Structural Deformations

Structural deformations or stresses from postulated accidents are limited to requirements contained in the ASME B&PV Code, Section III, Division 1, Appendix F, and the guidance per Section 4.2, Appendix A, of the SRP. Dynamic characteristics of the fuel assembly and grids were obtained from testing the assemblies for stiffness, natural frequencies and damping values, and used as inputs to analytical models for the fuel assembly and fuel channel. These tests were conducted with and without fuel channel. The test results, when compared with analysis results, have shown dynamic response of the ATRIUM 10XM design to be similar to other BWR fuel designs that have the same basic channel configuration and weight. The methodology for analysis of the channeled fuel assembly is described in References 16, 17, and 18. Evaluations of fuel under accident loadings include mechanical fracturing of the fuel rod cladding, assembly structural integrity, and fuel assembly liftoff. Table 3-2 of Enclosure 8 (in Reference 1) lists the margins for the fuel assembly components at the maximum acceleration allowed for the channel design.

Based on the above, the NRC staff reviewed the evaluation of the structural design of the assembly and fuel channel and found that the fuel assembly and channel meet all mechanical compatibility and strength requirements for use at MNGP.

3.2.4 ATRIUM 10XM Fuel Rod Thermal-Mechanical Evaluation

This section presents the results of the NRC staff's review of fuel rod thermal-mechanical (T-M) analyses for the ATRIUM 10XM fuel. The T-M analyses were performed using approved codes and methodology (References 9 and 12). The fuel cladding external oxidation limit was reduced according to a regulatory commitment made to the NRC when the RODEX4 code was first implemented (Reference 19). The RODEX4 fuel rod T-M analysis code and methodology are used to analyze the fuel rod for fuel centerline temperature, cladding strain, rod internal pressure, cladding collapse, cladding fatigue, and external oxidation.

Fuel Rod Design Evaluation

An ATRIUM 10XM fuel rod is slightly larger in diameter than the ATRIUM-10 fuel rod. The cladding and pellet dimensions were scaled according to the diameter change to maintain similarity in cladding strength. Since the diameter increase from the ATRIUM-10 design is considered small, the relative pellet-to-cladding gap size for the ATRIUM 10XM design is nearly the same as for the ATRIUM-10 design (Reference 22). In the ATRIUM 10XM fuel, the PLFRs are shorter in length than those in ATRIUM-10 and the fuel rods are made with Zircaloy-2 cladding that is cold-worked and stress relief annealed.

Table 2-1 of Enclosure 22 (in Reference 1) provides the main parameters for the fuel rod design evaluation results. In Enclosure 22 of Reference 1, Table 3-1 lists key fuel rod design parameters; Table 3-2 provides RODEX4 fuel rod results for equilibrium cycle conditions; and Table 3-4 lists cladding and cladding-end cap steady state stresses. The fuel rod analyses, such as those for fuel centerline temperature and cladding strain, cover normal operating conditions and AOOs.

Internal Hydriding

The absorption of hydrogen by the cladding can result in cladding failure due to reduced ductility and formation of hydride platelets. This is prevented by careful moisture control during fuel fabrication which reduces the potential for hydrogen absorption on the inside of the cladding.

Cladding Collapse

Creep collapse of the cladding and subsequent potential for fuel failure is avoided in the design by limiting the gap formation due to fuel densification subsequent to pellet-clad contact. Creep collapse of the clad is evaluated using RODEX4 (Reference 9). The RODEX4 code uses a statistical method and gives best-estimate results for nominal inputs. The maximum gap formation is calculated such that the expected fraction of fuel rods below the maximum value is 99.9 percent with a 95 percent confidence level.

Overheating of Fuel Pellets

To avoid fuel failure from overheating of the fuel pellet, the centerline temperature of the fuel pellets must remain below its melt temperature during normal operation and AOOs. The melting point is adjusted for gadolinia content in the fuel. AREVA establishes a linear heat generation rate (LHGR) to protect against fuel centerline melting during steady-state operation and during AOOs. Fuel centerline temperature is evaluated using the RODEX4 code (Reference 9) for both normal operating conditions and AOOs. The RODEX4 fuel model considers the fuel column divided into axial and radial regions, a gap region, cladding, gas plena, and the fill gas and released fission gases. Operational conditions are controlled by the **[[**

]]. The heat conduction for the clad and the fuel is calculated with a general variable mesh to accommodate steep temperature gradients.

Mechanical processes include **[[**

]].

Fuel rod power histories are generated and based on full-core symmetry to represent the fuel batch under evaluation. **[[**

]].

Uncertainty in the calculated channel bow leads to an associated uncertainty in the fuel rod power level. This uncertainty in power is taken into account as part of the RODEX4 statistical application methodology. A series of steps are then carried out to assess the effect of channel bow and its associated model uncertainty on the fuel rod thermal-mechanical behavior by accounting for channel bow in the generation of the fuel rod power histories. The MICROBURN–B2 code is used to model channel bow and assess the change in fuel rod power due to channel bow. **[[**

]].

[[

]].

Uncertainties that are taken into account in the analysis consist of **[[**

]]. The method covers the evaluation of fuel centerline temperature, cladding transient strain, cladding fatigue, cladding collapse, cladding external oxidation, and rod internal pressure.

Stress and Strain Limits

Cladding strain caused by transient-induced deformations of the cladding is calculated using the RODEX4 code and methodology, as described in Reference 9. The strain limit is reported to be less than 1 percent.

Cladding stresses are calculated using solid mechanics elasticity solutions and finite element methods. Stresses are calculated for the primary and secondary loadings. **[[**

]]. The stresses were found to be less than the design limits prescribed by Section III of the ASME B&PV Code.

Fuel Densification and Swelling

Fuel densification and swelling are limited by the design criteria for fuel temperature, cladding strain, cladding collapse, and rod internal pressure criteria.

Fatigue

Each fuel rod history is evaluated for power changes. The allowable number of cycles for every power change is determined from the cyclic stress calculated by the RODEX4 code along with a design fatigue S-N curve for zircaloy (Reference 15). A maximum value that encompasses 99.9 percent of the fuel rods with a 95 percent confidence is determined. The maximum cumulative usage factor (CUF) for the cladding remains below the design criterion.

Oxidation, Hydriding, and Crud Buildup

The RODEX4 calculation of cladding external oxidation includes an enhancement factor that is derived from poolside measurement data to obtain a fit of expected oxide thickness. An uncertainty on the model enhancement factor is also determined from this data. The RODEX4 analysis implicitly includes the thermal effect from normal levels of crud [deposits]. Specific analyses are performed for higher than normal crud deposition. An abnormal level of crud is defined by a formation that increases the calculated fuel average temperature by 25 degrees Centigrade (°C) above the design basis calculation. The corrosion model also takes into consideration the effect of the higher thermal resistance from the crud on the corrosion rate.

The highlights of a safety evaluation report (SER) restriction imposed on RODEX4 for crud are given below:

> *RODEX4 has no crud deposition model. Due to the potential impact of crud formation on heat transfer, fuel temperature, and related calculations, RODEX4 calculations must account for a design basis crud thickness. The level of deposited crud on the fuel rod surface should be based upon an upper bound of expected crud and may be based on plant-specific history. Specific analyses would be required if an abnormal crud or corrosion layer (beyond the design basis) is observed at any given plant. For the purpose of this evaluation, an abnormal crud/corrosion layer is defined by a formation that increases the*

> *calculated fuel average temperature by more than 25°C beyond the design basis calculation* **...**

As required by the SER restriction which defines an abnormal crud or corrosion layer that increases the calculated average temperature by 25°C (77°F) beyond the design basis calculation, analyses are already performed for each cycle. If plant specific measurements indicate abnormal levels of crud, then further analysis is performed for the plant using plant-specific data. If liftoff levels are found to be greater than those used in the RODEX4 corrosion model benchmark, then a plant-specific crud thickness will be input to encompass the total liftoff thickness. The term liftoff refers to the separation or liftoff of the eddy current measurement probe from the metallic surface of the fuel rod due to the presence of the insulating corrosion and crud layers. The crud input serves to satisfy the SER restriction on the design basis crud layer in cases where abnormal crud is encountered. A typical example was presented as part of the licensee's response to a staff's RAI (Reference 3), and illustrates the validity of the inherent assumption of thermal conductivity for normal liftoff. This example illustrates that a combined layer of oxide and crud includes the selection of a conservative oxide thermal conductivity that contributes to the composite thermal resistance.

At MNGP, the transition to ATRIUM 10XM design **[[**

]]. The licensee provided AREVA with a prior water chemistry evaluation that concludes the current water chemistry conditions at MNGP do not reduce fuel reliability margins through Cycle 28.

During the first reload application of RODEX4, the initial approved limit of oxide was challenged by the NRC staff based on a concern regarding the effect of spallation on cladding integrity. To avoid the issue of spallation, the oxide limit was reduced to **[[ ]]**. The **[[ ]]** limit was established from a review of historical liftoff measurement data on AREVA BWR fuel. This new limit was established, in part, as a means **[[ ]].** The NRC staff accepted the new fuel rod oxide limit, and thereby finds **[[ ]]** continued acceptable fuel performance.

Rod Internal Pressure

The fuel rod internal pressure is calculated using the RODEX4 code and methodology (Reference 9). The maximum rod pressure is calculated under both steady-state and transient conditions. Rod internal pressure is limited to **[[ ]]** above rated system pressure.

<u>Summary</u>

The NRC staff reviewed the licensee's application of approved code and methodologies in the fuel rod thermal-mechanical analyses for the AREVA ATRIUM 10XM fuel design that will be loaded and used for operation at MNGP. The staff determined that the fuel design criteria, as supported by the applicable regulations and sections of NUREG-0800, have been satisfied and provide reasonable assurance for safe operation at MNGP.

3.2.5 Thermal Hydraulic Design of ATRIUM 10XM Fuel Assemblies for MNGP

This section describes the NRC staff's evaluation of the MNGP thermal-hydraulic analyses to demonstrate the hydraulic compatibility of the ATRIUM 10XM fuel with co-resident fuel. NSPM is proposing to transition from the current GNF GE14 fuel design to AREVA ATRIUM 10XM fuel starting in Cycle 29 (i.e., spring of 2017). Enclosure 10 of Reference 1 provides the results of the thermal-hydraulic analyses to support that ATRIUM 10XM fuel is hydraulically compatible with the co-resident GE14 fuel. The results from the thermal-hydraulic analysis is compared to acceptance criteria established in NRC-approved topical reports ANF-89-98(P)(A), Revision 1, Supplement 1 (Reference 12) and XN-NF-80-19(P)(A), Volume 4, Revision 1 (Reference 20).

The thermal-hydraulic analyses were performed to verify that the design criteria were satisfied and further establish thermal operating limits with acceptable margins of safety during normal reactor operation and AOOs. Due to reactor and cycle operating differences, many of the analyses supporting these thermal-hydraulic operating limits were performed on a plant- and cycle-specific basis and are documented in plant- and cycle-specific reports (Reference 1). Table 3.1 of Enclosure 10 (in Reference 1) lists the applicable thermal-hydraulic design criteria, analyses, and results for hydraulic compatibility, thermal margin performance, fuel centerline temperature, rod bow, bypass flow, stability, LOCA analysis, CRDA analysis, ASME overpressurization analysis, and seismic/LOCA liftoff. The sections below summarize the results from selected design criteria and analyses results.

Hydraulic Characterization

Basic dimension parameters for ATRIUM 10XM and co-resident GE14 fuel designs are summarized in Table 3.2 of Enclosure 11 (AREVA Report ANP-3092(NP)) to Reference 1. Table 3.3 of Enclosure 11 provides loss coefficients that include modifications to the test data reduction process. The modifications account for **[[**

]]. The bare rod friction, ULTRAFLOW spacer, UTP, and LTP losses for AREVA ATRIUM 10XM fuel are based on tests performed by AREVA at its Portable Hydraulic Test Facility.

In its response to NRC staff SNPB RAI-5 (Reference 3), the licensee provided details of the test data reduction process mentioned above, and its modification to account for the flow through the LTP to the various leakage paths for the flow expansion differences between the test setup and the actual fuel assemblies. The irreversible pressure loss coefficient for the orifice and LTP is computed analytically using a model developed from test data. The resistance value for the flow through the LTP flow holes is derived analytically based on the size and the geometry of the holes. The small amount of leakage flow between the fuel support and the LTP is determined from information obtained either from GE/GNF or data from the co-resident fuel already present in the core.
The NRC staff finds that the introduction of ATRIUM 10XM fuel design does not significantly affect the hydraulic characterization for loss coefficients and pressure drops with a mixed core at MNGP during the transition to ATRIUM 10XM fuel design.

Thermal-Hydraulic Compatibility

The thermal-hydraulic compatibility analyses were performed in accordance with the AREVA thermal hydraulic methodology for BWRs (Reference 20). The XCOBRA code predicts the steady-state thermal hydraulic performance of fuel assemblies in BWR cores at various operating conditions and power distributions. The thermal-hydraulic compatibility analysis evaluates the relative thermal performance of the ATRIUM 10XM and GE14 fuel designs that will be inserted in the MNGP core. The analyses were performed for full-core GE14 and full-core ATRIUM 10XM configurations. The analyses for mixed-core configurations were also performed to demonstrate the thermal-hydraulic compatibility for resident and co-resident fuel designs.

A hydraulic compatibility analysis models each of the fuel assembly channels in the core such that the pressure drop across the channels is the same. Enclosure 11 (AREVA Report ANP-3092(NP)) to Reference 1, Table 3.4, lists a summary of all inputs, including core loading for representative transition cores, to the hydraulic compatibility analysis. The inputs are for rated (100 percent power / 100 percent flow) and off-rated (59.2 percent power / 43.3 percent flow) core thermal power and core flow conditions, in which the core flow follows the minimum pump speed on the MELLLA line. The off-rated statepoint was added to demonstrate that hydraulic compatibility is maintained for both rated and off-rated conditions. Analysis at two state points (at rated and off-rated conditions) was performed in the compatibility analyses to adequately support operation in the power-to-flow map. The selection of the off-rated state point considered the following operational conditions: (1) the lowest core flow at rated core power; (2) the highest core flow at rated core power; (3) core flow at the minimum pump speed (or the stability exclusion region) and highest core power (MEOD line, MELLLA line, etc.); and (4) the highest core flow at lowest allowable core power. The core flow at the minimum pump speed usually provides the most variation between fuel designs at off-rated conditions. For MNGP, this is represented at 59.2 percent power and 43.3 percent flow.

The evaluations were made for all transition core loading operational conditions (1) through (4), and with bottom-, middle-, and top-peaked axial power distributions as presented in Table 3.1 of ANP-3092(NP) (in Reference 1) and as illustrated in Figure 3.1 of ANP-3092(NP). Table 3.4 of ANP-3092(NP) provides a listing of all thermal-hydraulic design conditions, as well as the number of GE14 and ATRIUM 10XM assemblies for a full core GE14 loading; a first and second transition core loading, and a full core ATRIUM 10XM core loading. Tables 3.5 and 3.6 provide a summary of calculated thermal-hydraulic results for the transition core configurations. Tables 3.7 and 3.8 provide results for all the transition core configurations for both rated and off-rated power and flow conditions. As shown in Tables 3.7 and 3.8 for both rated and off-rated conditions, respectively, the flow to maximum power ATRIUM 10XM assembly is **[[**

]].

Based on the changes in pressure drop and assembly flow caused by the transition from GE14 fuel to ATRIUM 10XM fuel, the NRC staff finds that hydraulic compatibility analyses for the transition cores at MNGP provide reasonable assurance that the resident and co-resident fuel designs will satisfy the thermal-hydraulic design criteria for mixed cores.

Thermal Margin Performance

The thermal margin analyses were performed using thermal-hydraulic methodology and the XCOBRA code. The calculation of fuel assembly critical power ratio (CPR) (thermal margin performance) was established by means of an empirical correlation based on results of boiling transition test programs. The details of the CPR calculations are discussed in Section 2.1.1 of this SE. The CPR methodology for AREVA ATRIUM 10XM is the approach that was used by AREVA to determine the margin to thermal limits for BWRs, and the methodology is described in References 11 and 25. For the GE14 fuel, CPR values are calculated using the SPCB critical power correlation (Reference 21). Fuel assembly design features are incorporated in the CPR calculation through the K-factor term in the ACE correlation and the F-eff term for the SPCB correlation. The K-factors are based on the local power peaking from the nuclear design and are a function of void fraction and exposure. The additive constants are determined for each rod position based on critical power testing and calculated using NRC-approved methodologies (References 11 and 21).

NRC approved topical report EMF-2245(P)(A) (Reference 27) describes the processes for the application of approved BWR CP correlations (CPC) to the co-resident fuel remaining from prior cycles, similar to the situation projected for MNGP. The topical report describes two processes for applying the approved AREVA CPC to co-resident fuel: indirect correlation application (ICA) and direct correlation application (DCA). In its response to an NRC staff RAI requesting which process is used at MNGP, the licensee stated that the ICA process will be used to apply the SPCB correlation for the GE14 fuel with the same additive constants (Reference 3).

There are three steps listed in Reference 27. They are **[[**

]].

The compatibility between the ATRIUM 10XM and GE14 fuel designs has been evaluated at steady-state conditions with radial peaking factors (RPFs) between **[[ ]]** using representative K-factors and F-effs to provide relative CPR changes to determine the impact of ATRIUM 10XM fuel during the transition from GE14. Enclosure 11 (AREVA Report ANP-3092(NP)) to Reference 1, Tables 3.5 and 3.6, provide CPR results of transition cores for rated and off-rated conditions. Tables 3.7 and 3.8 show similar comparisons of CPR and assembly flow for transition core configurations beginning from full-core GE14 fuel to full-core ATRIUM 10XM fuel.

Based on the above, the NRC staff finds that the introduction of ATRIUM 10XM fuel will not cause an adverse impact on thermal margin for the co-resident fuel.

Rod Bow

The differential expansion between the fuel rods and cage structure, including lateral thermal and flux gradients, can lead to lateral creep bow of the rods in the spans between spacer grids. This lateral creep bow alters the pitch between the rods and may affect the peaking and local heat transfer. The design criteria related to rod bow is that **[[**
]].

AREVA, in response to an NRC staff RAI (Reference 3), provided a summary of procedures used to determine the impact of rod bow on thermal margin at lower and higher exposures of ATRIUM 10XM fuel at MNGP. **[[**

]]. Though minimum critical power ratio (MCPR) penalties are predicted for the ATRIUM 10XM design for exposures greater than **[[ ]]**, thermal margin is not expected to be impacted because of the lower reactivity of the fuel assembly at higher exposures**.**

Bypass Flow

The total core bypass flow is defined as leakage flow through the LTP flow holes, channel seal, core support plate, and LTP-fuel support interface. Enclosure 11 (AREVA Report ANP-3092(NP)) to Reference 1, Tables 3.7 and 3.8, provide results of core bypass flow fraction for the rated and off-rated conditions of power and core flow during the transition from full-core GE14 fuel to full-core ATRIUM 10XM fuel. The difference in bypass flow fractions between other transition core combinations of ATRIUM 10XM fuel and GE14 fuel are either equal to, or less than, the results from full-core GE14 fuel to full-core ATRIUM 10XM fuel.

Based on the above, the NRC staff finds that adequate bypass flow will be available with the introduction of the ATRIUM 10XM fuel design, and that applicable design criteria will be met.

<u>Summary</u>

The NRC staff reviewed all thermal-hydraulic analyses and results for demonstrating that the AREVA ATRIUM 10XM fuel design is hydraulically compatible with GE14 fuel for use at MNGP. The staff determined that the generic thermal-hydraulic design criteria, as approved by the NRC in topical report ANF-89-98(P)(A), Revision 1, and Supplement 1 (Reference 12) has been used in the analyses. The staff finds that although the ATRIUM 10XM and GE14 fuel assemblies are geometrically different, they remain hydraulically compatible.

3.2.6 MNGP Fuel Transition – Cycle 28 Fuel Cycle Design

Enclosure 15 (AREVA Report ANP-3215(NP)) to Reference 1 summarizes the fuel cycle design and fuel management calculations for ATRIUM 10XM at MNGP for a representative operating Cycle 28. These analyses were performed using the AREVA neutronic methodology, CASMO-4 lattice depletion code for generation of nuclear cross section data, and MICROBURN-B2 and the 3-dimensional core simulator code for pin power reconstruction for thermal margin analysis (Reference 24).

In the Cycle 28 representative core design, the reactor core contains **[[ ]]** fresh ATRIUM 10XM fuel assemblies with an average enrichment of slightly over **[[ ]]**. Appendix B, Figures B.1 through B.3 of ANP-3215(NP), provide Cycle 28 fresh reload fuel design axial enrichment and gadolinia distributions. The loading pattern maintains full core symmetry with the exception of some interior locations. Appendix A to ANP-3215(NP) shows acceptable power peaking and associated margins to limits for projected Cycle 28 operation. The specific core location of the fresh assemblies for Cycle 28 is provided in Appendix C to ANP-3215(NP).

Table A.1 of Appendix A to ANP-3215(NP) provides step-by-step depletion summary, control rod patterns and core average axial power and exposure distributions. Table A-1 of Appendix A lists operating parameters, including calculated $k_{eff,}$ [k-effective, a measure of the ability of a reactor to regenerate neutrons by the fission process] core power, inlet subcooling, core minimum CPR, core maximum LHGR, and core MAPLHGR for each incremental burnup step during fuel depletion. Table A.2 provides thermal margin calculation results for core limiting CPR, fraction of limiting CPR, core limiting LHGR, fraction of limiting LHGR, core limiting and fraction of APLHGR for depletion steps. Table 3.2 of ANP-3215(NP) presents hot operating target k_{eff} values at various cycle exposures and the k_{eff} and margin to limits from the design depletion analysis that are presented graphically in Figures 2.1 and 2.2 of ANP-3215(NP).

The equilibrium cycle design calculations have demonstrated adequate hot excess reactivity, standby liquid control (SLC) shutdown margin and cold shutdown margin throughout the cycle, as illustrated in Table 3.4 of ANP-3215(NP). The shutdown margin is in conformance with the Technical Specification limit of R + 0.38 percent Δk/k at BOC [beginning of cycle].

The licensee demonstrated that the hot excess reactivity and shutdown margin are maintained per technical specification values during the transition cycles and during Cycle 28 operation at MNGP. The design and licensing process requires that cycle exposure dependent hot and cold critical eigenvalues be selected for the design cycle of interest. Once the design eigenvalue bases of a cycle are established, and the core is designed and licensed, the site is provided with data to support the testing used to demonstrate compliance with the reactivity-related technical specifications.
Based on the above, the NRC staff finds that the cycle design calculations and projected control rod patterns for the equilibrium cycle are developed to be consistent with a conservative margin to thermal limits.

3.2.7 ACE/ATRIUM 10XM Critical Power Correlation

The ACE/ATRIUM 10XM CPC (Reference 11), as revised by Enclosure 13 (ANP-3138(NP)) to Reference 1, is used in the licensing analysis for MNGP. Reference 11 presents the approved ACE/ATRIUM 10XM CPC for the ATRIUM 10XM fuel design. Deficiencies were identified in the calculation of the K-factor within the ACE/ATRIUM-10XM CPR correlations (Reference 26), and these deficiencies were shown to have an influence on the predicted results in a non-conservative manner for this CPR correlation, for fuel assemblies with downskew axial power shape. Since K-factor was integrated over the entire heated length of the assembly, it was possible for the local peaking factors in the upper lattices to contribute significantly to the K-factor used, even when dryout occurs much lower in the bundle. The K-factor methodology was modified in response to deficiencies found in the axial averaging process. Also, the additive constants were revised as a result of the change to the K-factor model in Reference 11.

The critical power behavior of the individual fuel rods within the fuel bundle is influenced by the spaces and bundle geometry. Additive constants are factors that distinguish the critical power performance of each rod, are position dependent, and are considered as a flow/enthalpy redistribution characteristic for a given bundle and spacer design. The axial resolution of the model was increased to more accurately capture the shape of the axial power distribution for each rod in the assembly. With these revisions implemented, the additive constants were re-derived using the same procedure documented originally in ANP-10298PA, Revision 0 (Reference 41).

While reviewing the MNGP fuel transition LAR, the NRC staff completed its review of the Reference 26 topical report, which is a supplement to Reference 11. AREVA incorporated the accepted version of Reference 26 into the previously NRC-approved Reference 41, creating Reference 11. Reference 11 describes the ACE/ATRIUM 10 critical power correlation for BWR ATRIUM 10XM fuel design at MNGP. This correlation is designed for application to steady-state design analysis, core monitoring, transient AOOs, transient accidents, LOCAs, and instability analysis for the ATRIUM 10XM fuel design

Based on the above, the NRC staff finds that the overall effect of the changes documented in the improved K-factor methodology introduces physically realistic modeling of the local subchannel hydrodynamics that influence dryout behavior in a fuel rod array. The staff accepts the proposed corrections as acceptable improvements in the dryout modeling approach used in the ACE/ATRIUM-10XM CPR correlation that will be used in thermal margin calculations at MNGP.

3.3 Summary and Conclusions

The NRC staff has reviewed the licensee's application (Reference 1), in conjunction with the supplemental information (Reference 2) providing responses to the staff's requests for additional information and applicable methodologies to evaluate the acceptability of the MNGP transition to AREVA ATRIUM 10XM fuel with AREVA fuel performance assessment, safety analysis and core design methodologies.

Based on its review, the NRC staff determined that the licensee provided adequate technical basis to support the proposed TSs changes. The staff finds that the licensee has demonstrated that (1) MNGP complies with the staff limitations and conditions imposed for application of the topical reports, (2) AREVA codes and methods are applicable for MNGP, and (3) the proposed TSs changes are acceptable.

4.0 EVALUATION - RADIATION PROTECTION AND CONSEQUENCES BRANCH

4.1 Regulatory Evaluation

The NRC staff established the requirements and methodologies for evaluating the radiological consequences of the postulated DBAs using the dose criteria specified in 10 CFR 50.67, "Accident source term", and the guidance described in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." The requirements of 10 CFR 50.67 state that the applicable dose acceptance criteria are 5 rem Total Effective Dose Equivalent (TEDE) in the control room (CR), 25 rem TEDE at the exclusion area boundary (EAB), and 25 rem TEDE at the outer boundary of the low population zone (LPZ). The FHA-specific and CRDA-specific dose acceptance criteria are specified in SRP, Section 15.0.1, Revision 0, "Radiological Consequence Analyses Using Alternative Source Terms" (Reference 40). The dose acceptance criteria for the FHA and CRDA are a TEDE of 6.3 rem at the EAB for the worst 2 hours, 6.3 rem at the outer boundary of the LPZ, and 5 rem in the CR for the duration of the accident. RG 1.183 provides guidance to licensees on acceptable application of alternative source term (AST) submittals, including acceptable radiological analysis assumptions for use in conjunction with the accepted AST. The NRC staff also considered relevant information in the MNGP USAR), TSs, and applicable previous licensing actions for MNGP.

The original AST analyses for MNGP were submitted for NRC staff approval in a license amendment request dated September 15, 2005[10]. The submittal contained the radiological consequence analyses based upon the AST methodology for the following four DBAs that result in control room and offsite exposure.

- Loss of Coolant Accident (LOCA)
- Fuel Handling Accidents (FHA)
- Control Rod Drop Accident (CRDA)
- Main Steam Line Break (MSLB)

The submittal also included changes to the MNGP TSs and associated Bases to reflect implementation of AST assumptions in accordance with 10 CFR 50.67. By letter dated December 7, 2006[11], the NRC approved the AST methodology and the associated TSs as Amendment No. 148. By letter dated April 17, 2007[12], the NRC issued a correction to the safety evaluation associated with the MNGP AST amendment. The correction letter addressed typographical errors and did not change the NRC staff's conclusions regarding Amendment No. 148.

4.2 Technical Evaluation

A modification to the licensing basis fuel type can have the potential to change the core isotopic distribution assumed in post-accident conditions. Therefore, for the proposed amendment, the nuclide inventory of ATRIUM 10XM fuel must be evaluated versus the inventories in the AST analysis of record. To develop the core inventory used for the source term evaluation, the licensee used the ORIGEN isotope generation and depletion computer code which is consistent with NRC guidance.

LOCA

To support the proposed amendment, the licensee revised the LOCA radiological consequence analysis using the previously approved AST methodology which is described in MNGP USAR Section 14.7.2. The revised calculation uses the source terms for the proposed ATRIUM 10XM fuel design. All other methods and inputs that were approved in Amendment No. 148 to the MNGP remain unchanged. The licensee analysis resulted in an increase in the calculated dose as shown in Table 1 of this SE (see Page 39). The NRC staff reviewed the methods, parameters, and assumptions that the license used in its LOCA radiological dose consequence analysis and finds that that they are consistent with the guidance provided in RG 1.183. The NRC staff compared the doses estimated by the licensee and concluded that the radiological consequences at the EAB, LPZ, and in the CR are within the dose criteria specified in 10 CFR 50.67.

FHA

The FHA is described in MNGP USAR Section 14.7.6. The licensee reviewed the quantity of fuel rod damage following the postulated drop of an ATRIUM 10XM fuel assembly and calculated radiological source term of ATRUIUM 10XM fuel rod gases. The licensee's analysis

[10] ADAMS Accession No. ML052640366
[11] ADAMS Accession No. ML062850049
[12] ADAMS Accession No. ML070990089

estimated 162 fuel rod failures in the ATRIUM 10XM, which is fewer than the 172 fuel rod failures that was previously calculated and described in the MNGP USAR. The licensee's evaluation also showed that the overall accident dose from a FHA would be lower for the ATRIUM 10XM fuel than the GE14 fuel. Based on the above, the NRC staff finds that the current MNGP FHA analysis to be bounding. Therefore, the MNGP FHA regulatory dose limits are unaffected and still meet the regulatory requirement in 10 CFR 50.67 and the accident specific dose criteria described in SRP 15.0.1.

CRDA

To support the proposed amendment, the licensee revised the CRDA radiological consequence analysis using the previously approved AST methodology which is described in MNGP USAR Section 14.7.1. The licensee will evaluate the CRDA on a cycle-specific basis when using AREVA methods. The licensee performed an evaluation of the CRDA for a representative transition cycle. The evaluation showed the number of rods calculated to fail in this event remains below the value of 850 assumed in the MNGP USAR radiological evaluation of this event.

The revised CRDA calculation uses the source terms for the proposed ATRIUM 10XM fuel design. All other methods and inputs that were approved in Amendment No. 148 to the MNGP remain unchanged. The licensee analysis resulted in an increase in the calculated dose as shown in Table 1 of this SE. The NRC staff reviewed the methods, parameters, and assumptions that the licensee used in its CRDA radiological dose consequence analysis and finds that that they are consistent with the guidance provided in RG 1.183. The NRC staff reviewed the doses estimated by the licensee and concluded that the radiological consequences at the EAB, LPZ and in the CR are within the dose criteria specified in 10 CFR 50.67 and accident specific dose criteria described in SRP 15.0.1.

MSLB

The MSLB accident is described in MNGP USAR Section 14.7.3. As stated in the USAR, no fuel failures are expected to occur as a result of this accident. The radionuclide inventory released from the primary coolant system is present in the coolant prior to the event. Therefore, MSLB accident analysis is not affected by a change in fuel design. Based upon this information, the NRC staff finds that the proposed fuel design change does not alter the radiological consequences of a MSLB accident. The MNGP MSLB regulatory dose limits are unaffected and continues to meet the regulatory requirement in 10 CFR 50.67 and accident specific dose criteria described in SRP 15.0.1.

4.3 Summary and Conclusions

The NRC staff reviewed the analyses used by the licensee to assess the radiological impacts of the transition from GE14 fuel design to AREVA ATRIUM 10XM fuel design at MNGP. The staff finds that the licensee used methods consistent with regulatory requirements and guidance identified in Section 4.1 above. The staff also finds, with reasonable assurance that the licensee's estimates of the EAB, LPZ, and control room doses will continue to comply with these criteria. Therefore, the proposed change is acceptable with regard to the radiological consequences of postulated DBAs.

Table 1

MNGP Fuel Transition (at EPU) Accident Dose (in Rem TEDE)

	CRDA	LOCA	Regulatory Limit
EAB	2.01	1.47	25 (6.3 for CRDA)
LPZ	0.92	1.99	25 (6.3 for CRDA)
CR	1.89	3.83	5

5.0 TECHNICAL SPECIFICATION CHANGES

The licensee proposed changes to Appendix A, Technical Specifications, in order to implement its license amendment request. The technical bases for these changes have been evaluated by the NRC staff as set forth in the sections above. Therefore, the information provided below only describes and summarizes the proposed TS changes.

TS 2.1.1 "Reactor Core SLs [Safety Limits]"

The proposed change to TS 2.1, "SLs," revises the reactor core safety limits to reduce the value of the reactor steam dome pressure in TSs 2.1.1.1 and 2.1.1.2 from 785 psig to 586 psig. The reactor core safety limits are established to maintain the fuel cladding integrity and no significant fuel damage is calculated to occur if the safety limits are not exceeded.

The NRC approved License Amendment No. 185 for MNGP on November 25, 2014 (ADAMS Accession No. ML14281A318). This amendment resolved a 10 CFR Part 21 concerning a potential to momentarily violate the TS 2.1.1.1 SL during a pressure regulator failure maximum demand (open) transient. The value for TSs 2.1.1.1 and 2.1.1.2 were changed from 785 psig to 686 psig. This pressure value is applicable for the GEH safety analysis methodology.
In its November 11, 2014, letter (Reference 6), the licensee proposed modifying TSs 2.1.1.1 and 2.1.1.2 to reflect both continued operation of full-core GE14 fuel through Cycle 28, yet address the necessary change required to support this license amendment and use of AREVA safety analysis methodology. As such, TSs 2.1.1.1 and 2.1.1.2 will provide the reactor core SL values for both GEH and AREVA methodologies.

As summarized in Section 2.5 of this safety evaluation, the NRC staff finds this proposed change to be acceptable.

TS 4.2.1 "Fuel Assemblies"

The proposed change to TS 4.2.1 revises the present language stating that fuel assemblies shall consist of fuel rods and water rods, to fuel assemblies shall consist of fuel rods and water rods or channels. The proposed change includes water channels to the fuel assembly description. As discussed in Section 3.2.1 of this safety evaluation, the ATRIUM 10XM fuel design is comprised of a 10x10 array of fuel rods with a square internal water channel. In addition, these water channels are explicitly modeled in the safety analysis for the AREVA ATRIUM 10XM fuel. The modeling is explicit or implicit depending on the capability and applicable of the code.

Based on the above, the NRC staff finds the addition of the statement “or water channels” to the description of the fuel assembly to be acceptable

TS 5.6.3 “Core Operating Limits Report (COLR)”

The proposed change to TS 5.6.3 adds AREVA safety analysis methods to the references list contained in TS 5.6.3.b. The AREVA analytical methods and topical reports are those utilized to evaluate the fuel mechanical design, along with both cycle-dependent and independent safety analyses, to establish limits identified in the COLR.

As summarized in Section 2.5 of this safety evaluation, the NRC staff finds this proposed change to be acceptable.

6.0 STATE CONSULTATION

In accordance with the Commission’s regulations, the Minnesota State official was notified of the proposed issuance of the amendment on May 6, 2015. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to the use of facility components located within the restricted area as defined in 10 CFR Part 20 or changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding as published in the *Federal Register* on September 9, 2014 (79 FR 53460). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

8.0 CONCLUSION

The 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) there is reasonable assurance that 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.

9.0 REFERENCES

1. Letter from Mark A. Schimmel, NSPM, to the U.S. NRC Document Control Desk (DCD), “License Amendment Request for Transition to AREVA ATRIUM 10XM Fuel and AREVA Safety Analysis Methodology,” dated July 15, 2013 (ADAMS Accession Package No. ML13200A185).

2. Letter from Karen D. Fili, NSPM, to the U.S. NRC DCD, “AREVA ATRIUM 10XM Fuel Transition - Response to Request for Additional Information (TAC No. MF2479),” dated January 31, 2014 (ADAMS Accession Package No. ML14035A297).

3. Letter from Karen D. Fili, NSPM, to the U.S. NRC DCD, “AREVA ATRIUM 10XM Fuel Transition - Response to Request for Additional Information (TAC No. MF2479), dated March 12, 2014 (ADAMS Accession No. ML14077A291).

4. Letter from Karen D. Fili, NSPM, to the U.S. NRC DCD, “AREVA ATRIUM 10XM Fuel Transition - Supplement to Describe the Effect of Local Power Range Monitor Surveillance Grace Period (TAC No. MF2479), dated April 29, 2014 (ADAMS Accession No. ML14153A498).

5. Letter from Karen D. Fili, NSPM, to the U.S. NRC DCD, “AREVA ATRIUM 10XM Fuel Transition - Response to Request for Additional Information (TAC No. MF2479),” dated May 9, 2014 (ADAMS Accession Package No. ML14132A189).

6. Letter from Karen D. Fili, NSPM, to the U.S. NRC DCD, “AREVA ATRIUM 10XM Fuel Transition License Amendment Request Supplement (TAC No. MF2479), dated November 11, 2014 (ADAMS Accession No. ML14323A026).

7. Letter from U.S. NRC, “Monticello Nuclear Generating Plant - Issuance of Amendment No. 176 to Renewed Facility Operating License Regarding Extended Power Uprate,” Docket No. 50-263, dated December 9, 2013 (ADAMS Accession No. ML13316C459).

8. Letter from U.S. NRC, “Monticello Nuclear Generating Plant - Issuance of Amendment No. 180 to Renewed Facility Operating License Regarding Maximum Extended Load Line Limit Analysis Plus, Docket No. 50-263, dated March 28, 2014 (ADAMS Accession No. ML14035A248).

9. AREVA NP, Inc., BAW-10247PA Revision 0, “Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors,” dated April 2008 (ADAMS Accession Nos. ML081340383 and ML081340385 (proprietary) and ML081340208 (non-proprietary)).

10. AREVA NP Inc., ANP-3138(P), Revision 0, “Monticello Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation,” dated August 2012 (ADAMS Accession Nos. ML13200A204 (proprietary) and ML13200A193 (non-proprietary)).

11. AREVA NP Inc., ANP-10298P-A, Revision 1, “ACE/ATRIUM 10XM Critical Power Correlation,” dated March 2014 (ADAMS Accession Nos. ML14183A739 (Part 1 of 3), ML14183A743 (Part 2 of 3), and ML14183A748 (Part 3 of 3) (proprietary) and ML14183A734 (non-proprietary)).

12. Advanced Nuclear Fuels Corporation, ANF-89-98(P)(A), Revision 1 and Supplement 1, “Generic Mechanical Design Criteria for BWR Designs,” dated May 1995 (ADAMS Accession No. ML081350281) (proprietary - no publicly-available copy located in ADAMS).

13. U.S. Nuclear Regulatory Commission, NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” dated March 2007 (ADAMS Accession No. ML070550060).

14. ASME Boiler and Pressure Vessel Code, Section III, Division 1, American Society of Mechanical Engineers.

15. W. J. O’Donnell and B. F. Langer, “Fatigue Design Basis for Zircaloy Components,” Nuclear Science and Engineering, Volume 20, dated January 1964 (no publicly-available copy located in ADAMS).

16. Framatome ANP Inc., EMF-93-177(P)(A) Revision 1, “Mechanical Design for BWR Fuel Channels,” dated August 2005 (no publicly-available copy located in ADAMS).

17. Exxon Nuclear Company, XN-NF-81-51(P)(A), “LOCA - Seismic Structural Response of an Exxon Nuclear Company BWR Jet Pump Fuel Assembly,” dated May 1986 (ADAMS Accession No. ML081720316) (proprietary - no publicly-available copy located in ADAMS).

18. Exxon Nuclear Company, XN-NF-84-97(P)(A), “LOCA - Seismic Structural Response of an ENC 9x9 BWR Jet Pump Fuel Assembly,” dated August 1986.

19. Letter from Farideh E. Saba (NRC) to Michael J. Annacone (CP&L), “Brunswick Steam Electric Plant, Units 1 and 2 – Issuance of Amendments Regarding Addition of Analytical Methodology Topical Reports to Technical Specification 5.6.5 (TAC Nos. ME3858 and ME3859),” dated April 8, 2011 (ADAMS Accession No. ML11101A043).

20. Exxon Nuclear Company, XN-NF-80-19(P)(A), Volume 4, Revision 1, “Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads,” dated June 1986 (ADAMS Accession No. ML081700491) (proprietary - no publicly-available copy located in ADAMS).

21. AREVA, EMF-2209(P)(A), Revision 3, “SPCB Critical Power Correlation,” dated September 2009 (ADAMS Accession Nos. ML093650230 (proprietary) and ML093650235 (non-proprietary)).

22. Siemens Power Corporation, EMF-95-52(P), “Fuel Design Evaluation for Siemens Power Corporation ATRIUM-10 BWR Reload Fuel,” dated December 1998 (no publicly-available copy located in ADAMS).

23. Exxon Nuclear Company, XN-NF-75-32(P)(A), Supplements 1 through 4, “Computational Procedure for Evaluating Fuel Rod Bowing,” dated October 1983 (ADAMS Accession No. ML081710709) (proprietary - no publicly-available copy located in ADAMS).

24. Siemens Power Corporation, EMF-2158(P)(A) (proprietary), EMF-2158(NP)(A) (non-proprietary), Revision 0, “Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4 / MICROBURN-B2,” dated October 1999 (ADAMS Accession Nos. ML003698553 (proprietary) and ML003698495 (non-proprietary)).

25. AREVA NP Inc., ANP-10307PA, Revision 0, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," dated June 2011 (ADAMS Accession Nos. ML11259A022 (proprietary) and ML11259A021 (non-proprietary)).

26. AREVA NP Inc., ANP-10298PA, Revision 0, Supplement 1P, Revision 0, "Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation," dated December 2011 (ADAMS Accession Nos. ML11363A123 (proprietary) and ML11363A122 (non-proprietary)).

27. Siemens Power Corporation, EMF-2245(P)(A), Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-resident Fuel," dated August 2000 (ADAMS Accession Nos. ML003753223 (proprietary) and ML003753200 (non-proprietary)).

28. Advanced Nuclear Fuels Corporation, ANF-913(P)(A), Volume 1, Revision 1 and Supplements 2, 3, and 4, Richland, WA, "CONTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," dated August 1990 (ADAMS Accession Package No. ML081340221) (proprietary - no publicly-available copy located in ADAMS).

29. Letter from Pedro Salas, AREVA NP, Inc., to U.S. NRC DCD, "Response to NRC Letter Regarding Nuclear Fuel Thermal Conductivity Degradation Evaluation for Light Water Reactors Using AREVA Codes and Methods," dated April 27, 2012 (ADAMS Accession No. ML121220377 (letter) and ML121220368 (package)).

30. U.S. Nuclear Regulatory Commission, NUREG-1433, "Standard Technical Specifications - General Electric Plants BWR/4 Plants," Revision 4.0, Volume 2: Bases (ADAMS Accession No. ML12104A193).

31. Letter from Timothy J. O'Connor, NSPM, to U.S. NRC DCD, "Monticello Extended Power Uprate: Response to NRC Reactor Systems Branch and Nuclear Performance & Code Branch Request for Additional Information (RAI) dated February 23, 2009 (TAC No. MD9990)," Docket 50-263, dated April 22, 2009 (ADAMS Accession No. ML091130634).

32. Exxon Nuclear Company, XN-NF-80-19(P)(A), Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, (ADAMS Accession No. ML081750347) (proprietary - no public document available).

33. Letter from Terry A. Beltz, U.S. NRC, to Karen D. Fili, NSPM, "Monticello Nuclear Generating Plant – Issuance of Amendment to Revise the Technical Specifications to Support Fuel Storage System Changes (TAC No. MF9893)," dated October 24, 2014 (ADAMS Accession No. ML14197A020

34. Exxon Nuclear Company, XN-NF-80-19(P)(A), Volume 3, Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors - THERMEX: Thermal Limits Methodology Summary Description," dated January 1987 (ADAMS Accession No. ML081340305) (proprietary - no publicly-available copy located in ADAMS).

35. Exxon Nuclear Company, XN-NF-84-105(P)(A), "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," dated February 1987 (ADAMS Accession No. ML081340188) (proprietary - no publicly-available copy located in ADAMS).

36. Letter from Global Nuclear Fuel-Americas to U.S. NRC DCD, "Supplemental Reload Licensing Report for Monticello Reload 26 Cycle 27 Extended Power Uprate (EPU)," Docket 50-263, dated January 2013 (ADAMS Accession No. ML13191A569).

37. General Electric Nuclear Energy, NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology and Reload Applications," dated August 1996 (ADAMS Accession No. ML14093A210).

38. AREVA NP, Inc., BAW-10255PA, Revision 2, "Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code," dated May 2008 (ADAMS Accession No. ML082820002) (proprietary - no publicly-available copy located in ADAMS).

39. Siemens Power Corporation, EMF-CC-074(P)(A), Volume 4, Revision 0, "BWR Stability Analysis Assessment of STAIF with Input from MICROBURN-B2," dated August 2000 (ADAMS Accession No. ML090750216) (proprietary - no publicly-available copy located in ADAMS).

40. U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, Chapter 15.0.1, Revision 0, "Radiological Consequence Analyses Using Alternative Source Term," dated July 2000 (ADAMS Accession No. ML003734190).

41. AREVA NP Inc., ANP-10298P-A, Revision 0, "ACE/ATRIUM 10XM Critical Power Correlation," dated March 2010 (ADAMS Accession Nos. ML101190045 (proprietary) and ML101190044 (non-proprietary)).

42. Letter from Karen D. Fili, NSPM, to the U.S. NRC DCD, "AREVA ATRIUM 10XM Fuel Transition – Errata Regarding License Amendment Request (TAC No. MF2479)," dated May 9, 2014 (ADAMS Accession Package No. ML14132A197).

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